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The NEA Co-operative Programme on Decommissioning

A Decade of Progress

NEA No. 6185

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.

FOREWORD

The NEA Co-operative Programme for the Exchange of Scientific and Technical Information Concerning Nuclear Installation Decommissioning Projects (CPD) is a joint undertaking among member country organisations actively executing or planning the decommissioning of nuclear facilities. Initiated in 1985, the CPD recently completed 20 years of operation.

The objective of the CPD is to acquire information from operational experience in conducting specific decommissioning projects that is useful for future projects. Its working method is based on the exchange of knowledge currently drawn from 42 specific decommissioning projects. Such information includes, but is not limited to, project descriptions and plans; data obtained from research and development associated with decommissioning projects; and data and lessons learnt resulting from the execution of a decommissioning project.

Although some of the information exchanged within the CPD is confidential in nature and is restricted to programme participants, experience of general interest gained under the programme's auspices is released for broader use. Such information is brought to the attention of all NEA members through regular reports to the NEA Radioactive Waste Management Committee (RWMC).

This report, prepared by the CPD, describes the progress and generic results obtained by the Co-operative Programme on Decommissioning during the period 1995-2005. It follows a similar status report published by the NEA in 1996 covering the first ten years of the CPD.[1] The CPD's report has been brought to the attention of the RWMC Working Party on Decommissioning and Dismantling (WPDD), which found the information presented by the CPD valuable for all NEA member countries and therefore decided to publish this report to encourage other member countries and decommissioning projects to consider joining the CPD.

The RWMC and its Working Party on Decommissioning and Dismantling would like to thank the CPD for sharing the experience from its important work.

1. NEA (1996), *The NEA Co-operative Programme on Decommissioning: The First Ten Years, 1985-1995*, OECD/NEA, Paris.

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SUMMARY

The Co-operative Programme for the Exchange of Scientific and Technical Information Concerning Nuclear Installation Decommissioning (CPD) is a joint undertaking according to Article 5 of the Statute of the NEA. Concluded in 1985, the Agreement of the 23 participating organisations constituing the CPD has been continuously extended with the current programme period lasting until 2009. This report provides information about the participants, structure and achievements of the Co-operative Programme.

The objective of the CPD is to acquire information and share operational experience from the conduct of 42 current decommissioning projects, such as project description and design, data resulting from the execution of decommissioning projects, and associated research and development results. The information generated in the project is protected by confidentiality provisions, which allow for a frank and open exchange of experiences, on a "give and take" basis. The information exchange also ensures that the best internationally available experience is available and that safe, environmentally friendly and cost effective methods are employed in all decommissioning projects.

The Co-operative Programme is implemented by a Management Board (MB) representing the participating organisations and a Technical Advisory Group (TAG) for the information exchange between the individual decommissioning projects. The latter benefits from the support of a Programme Co-ordinator financed by the Programme.

The projects in the Programme have a broad range of characteristics and cover various types of reactors and fuel facilities. The Programme now covers 26 reactors, 8 reprocessing plants and 8 fuel facility projects, representing a wide selection of facility types in each category. Also, all three stages of decommissioning – from active dismantling to safestore and to completed decommissioning back to "green field conditions" – are represented.

Over the 20 years of experience of the Co-operative Programme on Decommissioning, and in particular through the information exchange and review within the TAG, it has become evident that:

- decommissioning can and has been done in a safe, cost-effective and environmentally friendly manner;
- current technologies have demonstrated their effectiveness and robust performance in numerous decommissioning activities; and
- feedback of experience on design, construction and operation is a considerable help for reliable planning, cost evaluation and successful realisation of a decommissioning project.

Regarding technical challenges, specific trends have been observed over the last decade. Large contaminated components, for example heat exchangers, steam generators, large tanks etc., that have been segmented *in situ* into smaller pieces, are increasingly removed "in one piece" and transported outside the contained area into separated facilities for further processing. Regarding the use of robotics, the CPD observed that industrial robots may have a limited applicability in decom-missioning, in contrary to earlier expectations that robotic methods would be extensively used in the

dismantling of radioactive components, especially in the high radiation areas in fuel facilities. Experience collected within the CPD also pointed to challenges in the release of alpha contaminated areas, where seepage of contamination into cracks and reappearance of activity in walls previously declared as “clean” posed specific problems.

On the side of organisational trends a movement towards sequential licensing has been observed. This is seen as being advantageous for the management of projects, but also increases the efforts needed for documentation. Other challenges for management raise from company reorganisation, privatisation and budgetary difficulties.

The lessons learnt by the participants in the CPD have been helpful for individual projects in making project decisions and in many cases have influenced general project directions. Key examples concern decommissioning techniques like dry abrasive blasting, cutting, removal of biological shielding and decontamination of concrete surfaces. Regarding the use of ventilated suits for workers in specific decommissioning activities, the information exchange in the CPD has helped push programmes to either improve the ventilated suits or to reduce the need for these systems.

To address more general issues of common interest the CPD Technical Advisory Group established specific Task Groups. The Task Group on Decommissioning Costs developed, in co-operation with IAEA and EC, a standardised list of items for costing purposes, which allows for comparison of project costs. The Task Group on Recycling and Re-use put together case studies and provided information on the practicality and usefulness of the criteria under development for the release of slightly contaminated material from decommissioning, seen from the perspective of organisations engaged in actual decommissioning operations. Decontamination issues have been addressed by another Task Group that surveyed applied decommissioning techniques and characterised selected techniques in connection with decommissioning. Finally, the Task Group on Release Measurements studied collected relevant data from CPD projects and produced a report giving an overview on equipment and methods for and applications of release measurement.

Looking back over the twenty years since the OECD Nuclear Energy Agency established the CPD, the Programme has functioned as the main international forum for the exchange of technical and other information arising from nuclear decommissioning projects. In addition to the tangible benefits listed above, personal interaction among experienced people from a wide cross-section of the decommissioning community is also a valuable asset in itself.

As for the future, it is foreseen that participation in the Co-operative Programme will continue to grow and new projects are welcome to enter the information exchange.

1. INTRODUCTION

The Co-operative Programme for the Exchange of Scientific and Technical Information Concerning Nuclear Installation Decommissioning has recently completed 20 years of operation. The Programme, which is also known under the short title *Co-operative Programme on Decommissioning* (CPD), was established in 1985 as a joint undertaking according to Article 5 of the Statute of the OECD Nuclear Energy Agency (NEA). Based on a specific Agreement between organisations actively executing, planning or having plans regarding decommissioning of nuclear facilities, the CPD has grown from an initial 10 decommissioning projects from 9 participating organisations to 42 decommissioning projects from 23 organisations today.

The objective of the CPD is to acquire information from operational experience in decommissioning nuclear installations that is useful for future projects. Such information can be effectively obtained, collected and analysed through the sharing of scientific and technical knowledge drawn from the participating decommissioning projects. The information exchange also ensures that the best internationally experience is available and that safe, environmentally friendly and cost effective methods are employed in all decommissioning projects.

The roots of the Co-operative Programme on Decommissioning can be traced back to the late seventies, when the NEA started to explore the potential for international co-operative ventures in this area and organised (or sponsored) a number of technical workshops and conferences in the years 1978-1984. The NEA Radioactive Waste Management Committee (RWMC) sponsored, during 1981-1984, a survey over the status of decommissioning projects in member organisations and the needs of technology exchange in this field. The results were compiled in a report by the United State Department of Energy (USDOE) and UNC Nuclear Industries (United States) [1]. Another important result of the early NEA activities was the Ågesta Decontamination Project (1981-1982), where an international team from Germany, Italy, Sweden, Switzerland, the United Kingdom and the United States compared PWR decontamination methods. These NEA activities created a climate favourable to international co-operation in this field. So, when the USDOE proposed in 1984 the setting up, under the NEA, of a broad co-operation for the exchange of technical information between major decommissioning projects in member organisations, it rapidly obtained strong support and the Co-operative Programme could be launched in September 1985.

The initial Agreement inaugurating the Co-operative Programme on Decommissioning had been concluded for a five years term and was renewed in the years 1990, 1995 and 2000. However, in the year 2003 the agreement was revised, modernised and extended to include a financial mechanism to support the work of a Programme-Co-ordinator and a new Agreement was concluded to become into effect in 2004, again for a five years period until the end of 2008.

In principle, the CPD is a programme for decommissioning projects from NEA member countries. However, in specific cases and with notification to the NEA Steering Committee for Nuclear Energy, decommissioning projects from outside the OECD have been admitted to the programme. This has been in order to ensure and demonstrate that the best internationally experience is available to all nuclear decommissioning projects and that safe, environmentally friendly and cost effective methods are employed in all decommissioning projects.

Within the NEA, the Co-operative Programme on Decommissioning is linked to the Radioactive Waste Management Committee. The Programme itself is governed by a Management Board (the successor to the former Liaison Committee), where all participating organisations are represented. The main forum for the exchange of information is the Technical Advisory Group (TAG), which meets twice a year. A key factor in the success and utility of this programme has been the unanimous agreement of participants that information and experience exchange is based on a "give and take" approach. In addition to the TAG itself, Task Groups have been set up to study specific topics of common interest.

The Programme has published two overview five-year reports earlier [2,3]. This report concentrates on the Programme's development over last ten years and describes the activities of the Management Board, the Technical Advisory Group and the various Task Groups during that period. It shows how the Programme and its activities and procedures have evolved over the years and indicates the directions of developments in the organisation and execution of decommissioning projects. Finally, it will briefly overview the achievements of the Co-operative Programme and visualise future developments in the field. A description of the participating projects complements the description of the CPD's development.

2. STRUCTURE OF THE CO-OPERATIVE PROGRAMME

This chapter gives an overview of the structure of the CPD under the new Agreement and of how the various functions of the Programme and within the Programme have gradually evolved over the years.

2.1 Participating organisations

The NEA Co-operative Programme on Decommissioning functions within the framework of an agreement between a number of organisations actively executing, planning or with future plans regarding decommissioning of nuclear facilities. Currently, 23 organisations from Belgium, Canada, Chinese Taipei, France, Germany, Italy, Japan, Korea, Slovak Republic, Spain, Sweden and the United Kingdom are participants of the Agreement of the Co-operation Programme on Decommissioning. Due to changes in some projects and mergers of organisations the number and identity of participating organisations has changed over time. In earlier periods of the Programme, organisations from the United States and Estonia also participated in the CPD activities. A full list of current and former participants in the Co-operative Programme on Decommissioning is given in Table 1.

In addition to these organisations, which are Parties to the CPD Agreement, experts from international organisations, whose interest in decommissioning is of a more general nature, such as the International Atomic Energy Agency (IAEA), the European Commission (EC) and the International Union of Producers and Distributors of Electrical Energy (EURELECTRIC, formerly UNIPEDE), are frequently invited to participate as observers, to give and receive general and overview information on programmes, projects and task group activities. Also, on a case by case basis, there are a number of organisations supporting the Programme by assigning specialists to the various task groups and special arrangements.

2.2 Management Board

The Co-operative Programme is implemented by two groups: a governing body and a technical group. The governing body, called the Management Board (MB, formerly called Liaison Committee [LC]), comprises representatives of all participating organisations. It is responsible for the general conduct and orientation of the programme, including direction and supervision of the work programme, establishment of criteria for disseminating the information exchange generated within the Programme, approval of changes of membership, etc. The Secretariat of the Management Board is ensured directly by the NEA Secretariat. The MB meets once a year. A Management Board Bureau has been set up to generally oversee the working of the Programme and to take the near-term operational decisions necessary for the Programme to function satisfactorily.

2.3 Technical Advisory Group

The central forum for the exchange of information is the Technical Advisory Group (TAG), which meets twice a year, generally at the site of a participating project. It is composed of technical managers and other senior specialists from the currently 42 member projects in the Programme which include the decommissioning of 26 reactors, 8 reprocessing plants, 7 fuel material plants, and 1 isotope handling facility. (A full list of the projects currently participating in the Co-operative Programme is provided in Chapter 3 and more detailed information on the projects is given in Appendix 1.)

Table 1. **Organisations participating in the Programme**

Country	Organisation
Belgium	• Belgoprocess NV • Centre d'étude de l'énergie nucléaire/Studiecentrum vor Kernergie (CEN•SCK)
Canada	• Atomic Energy of Canada Limited/ Énergie atomique du Canada limitée (AECL/EACL)
Chinese Taipei	• Institute of Nuclear Energy Research (INER)
France	• AREVA NC[1] • CODEM GIE • Commissariat à l'énergie atomique (CEA) • Électricité de France (EDF)
Germany	• Arbeitsgemeinschaft Versuchsreaktor GmbH (AVR) • Energiewerke Nord GmbH (EWN) • Forschungszentrum Karlsruhe GmbH (FZK) • Wiederaufarbeitungsanlage Karlsruhe GmbH (WAK)
Italy	• Società Gestione Impianti Nucleari SpA (SOGIN)
Japan	• Japan Atomic Energy Agency[2] (JAEA) • Japan Atomic Power Co. (JAPCO) • The Radioactive Waste Management and Nuclear Facility Decommissioning Technology Center (RANDEC)
Republic of Korea	• Korea Atomic Energy Research Institute (KAERI)
Slovak Republic	• Slovenské Elektrarne A.S. (SE-VYZ Bohunice)
Spain	• Centro de Investigaciones Energéticas, Medioambientales Tecnológicas (CIEMAT) • Empresa Nacional de Residuos Radioactivos SA (ENRESA)
Sweden	• Svensk Kärnbränslehantering AB (SKB)
United Kingdom	• British Nuclear Fuels PLC (BNFL) • United Kingdom Atomic Energy Agency (UKAEA)
Former member organisations of the CPD	
	• Alara A.S. (Estonia)
	• Kernkraftwerk Lingen GmbH (Germany)
	• Comitato Nazionale per la Ricerca e per lo Sviluoppo dell'Energia Nucleare e delle Energi Alternative (ENEA) (Italy)
	• Department of Energy (USDOE) (USA) • Public Services of Colorado (USA)

As the Programme Agreement contains provisions and conditions protecting the information exchanged, TAG discussions are open and based on confidentiality among programme members. Because of this frank and open nature of the discussions, questions and answers at the TAG meetings,

1. The former COGEMA.
2. Before their merge into the Japan Atomic Energy Agency (JAEA), both Japan Atomic Energy Research Institute (JAERI) and Japan Nuclear Cycle Development Institute (JNC) participated in the CPD.

the practice has lately been developed to e-mail the draft summary record of each meeting to the participants in order to make sure that the record is accurate as well as to ensure that information they consider as restricted continues to be protected.

With the increase in the number of projects, the planning of TAG meetings has been streamlined for effectiveness. A meeting form that each TAG meeting participant fills in and mails to the meeting's hosting organisation, simplifies the organisation of TAG meetings. Projects are required to signal in advance the time they need for presentations, whether they intend showing videos, etc. The main purpose of the Programme being the exchange of information, the time for presentations and discussions is not limited. Experience has shown that (almost) three full days are necessary for reports from the projects and the task groups as well as the ensuing discussions.

As an informal method for widening the flow of information to the Co-operative Programme, some TAG meetings have had invited lecturers on special subjects. Some examples are Dr. Ingemar Lund's speech on "Regulatory Aspects of Decommissioning: A Regulator's Views" at TAG 22, Karlsruhe, Germany, May 1997, Dr. Michael Segal's speech on "Communication with the Public on Risks and Radiation" at TAG 25, Dessel, Belgium, November 1998 and Mr. Thomas LaGuardia's speech on "Commercial Decommissioning Programmes in the US" at TAG 26, Rome, Italy, April 1999.

The increase in the number of projects participating in the Co-operative Programme has led to a significantly broader exchange of scientific and technical information. During TAG meetings, each project gives its experience on the techniques and processes used (strategy, technique or process employed). This is subject to serious discussion with alternative suggestions being put forward whenever such successful experience is available. However, with the time constraints placed on the TAG meeting due to the number of projects wishing to participate in the exchange, further discussions take place at a later date outside the meeting. Some examples of this international exchange of information and experience are as follows:

1. During the decommissioning of AT 1, a pilot reprocessing facility for fast breeder's reactors spent fuel in France, significant experience was gained among others on alpha penetration of concrete and characterisation. This was seen as being very beneficial to the JRTF Japanese decommissioning projects. During one year in the 1990s, a Japanese colleague stayed as a trainee in the preparation of JRTF decommissioning.
2. The WAGR project has made significant use of the information and experience available from TAG meetings in such areas as the decontamination techniques to be used resolving operational problems in the fume filtration system. Also experience in radiological information was used in defining strategy and planning of decommissioning operations. In addition, information relating to waste management and volume reduction has proved beneficial.
3. Following TAG 35 held in Ottawa, the representative from Belgoprocess was asked to stay on at Chalk River to give further details on the work and techniques used on their decommissioning projects. This information and experience led to a change in the approach to the decommissioning of the old fuel reprocessing facility at Chalk River. In addition, the Belgian experience in the recycling of concrete launched a similar review within AECL. Further the Swedish approach in the use of *in situ* object counting system resulted in AECL purchasing two units for work on decommissioning in Whiteshell and Chalk River.
4. Following TAG 36 and the presentation on the restructuring of the UK decommissioning programme, the formation of the Liabilities Management Unit (LMU) and the Government assuming liability for legacy facilities, a presentation was made to AECL management. This was important to the Canadian Government, which later created a similar Liabilities Management Unit within the AECL.

2.4 Task Groups

It was apparent, after the first few meetings of the Technical Advisory Group, that there were a number of specific issues of general interest that required in depth concentrated analyses for which the Technical Advisory Group was not the most suitable forum. This was both due to the practical time limit for the Technical Advisory Group meetings and to the fact that such issues required the work of specialists. Special groups (Task Groups) were therefore established for conducting such studies/analyses.

Because the Co-operative Programme is basically a volunteer activity, participation in a Task Group often requires from members considerable volumes of work extra to their normal duties. The members of Task Groups must therefore have a deep commitment to the aim of the Task Group and the Co-operative Programme.

Task groups have worked in the following areas:

- Decommissioning costs.
- Recycling and re-use of slightly contaminated material from decommissioning.
- Decontamination in connection with decommissioning.
- Release measurements.

2.5 Special arrangements

One feature of the Agreement is the possibility to establish co-operative arrangements between two or more of the participants in the Programme.

Currently, a special arrangements project is underway for validating, on a fairly large scale, certain calculation programmes used nationally and internationally in the calculation of radiation doses from exposure to contaminated material during the recycling of steel. The radiation dose to workers will be measured during the processes of segmenting and melting radioactively contaminated scrap and then using the resulting ingots in the manufacture of rolls for metal industries. These actually measured doses will be compared to values calculated for the same processes using specific US and French codes. The project was initiated and is managed by the Swedish Radiation Protection Authority (SSI), the other participants being the USDOE, CEA (France), Argonne National Laboratory (USA), Studsvik AB (Sweden), Belgoprocess n.v. (Belgium) and Åkers AB (Sweden).

Previously, there have been such arrangements in progress between:

- JAERI and UKAEA.
- JAERI and CEA.

2.6 Programme co-ordinator

From the start of the Co-operative Programme, the smooth functioning of its arrangements and procedures had been ensured by the appointment of a Programme Co-ordinator. Hitherto, these services have been provided by Sweden (SKB). The Programme Co-ordinator basically acts as the secretariat for the TAG, supports the work of the CPDMB and co-ordinates every day work with the NEA Secretariat. However, SKB has during the last five-year programme decided to reduce its support to the Co-operative Programme and hence also the provision of the Programme Co-ordinator. The participants of both the TAG and the Liaison Committee concluded that for the continued efficient running of the Co-operative Programme, there is a need for the continued role undertaken by the Programme Co-ordinator. Since the participants of the Co-operative Programme were not able to provide for the required replacement Programme Co-ordinator, it was decided to ask each participant in the Programme to contribute to funding for the provision of a Programme Co-ordinator. This was duly agreed by the LC participants and so, since 2001, a financial contribution from each member in the CPD is sought.

2.7 Link to the NEA committee structure

Within the NEA committee structure decommissioning is linked to the Radioactive Waste Management Committee (RWMC). The RWMC has long recognised that decommissioning and waste management are intimately related and that decommissioning has a bearing on waste management and waste management influences decommissioning.

The RWMC created in the year 2000 the Working Party on Management of Materials from Decommissioning and Dismantling (WPDD) as its main support group to keep under review the policy, strategic and regulatory aspects of nuclear decommissioning. The WPDD is constituted of senior representatives of national organisations who, in their capacity as regulators, implementers, R&D experts or policy makers, have responsibility, broad overview and experience in the field.

Within the NEA, the Co-operative Programme on Decommissioning is linked and reports to the RWMC. The members of CPD, and the TAG, are all decommissioning implementers.

It is important that there is a high degree of co-operation and cross fertilisation between the WPDD and the CPD. To ensure this close relationship some delegates are members of both committees and bureaux, and co-operation between both groups has been formalised by an interface document.

3. PROGRAMME ACTIVITIES

3.1 Projects participating in the information exchange

As of November 2006, 42 individual decommissioning projects are participating in the information exchange of the Co-operative Programme on Decommissioning. They cover 26 reactor decommissioning projects and 16 projects on decommissioning of fuel cycle facilities.

The projects in the Programme have a broad range of characteristics and cover various types of reactors and fuel facilities. A full list of the projects that are or have been participating in the information exchange of the CPD is given in Table 2 (reactor projects) and Table 3 (fuel cycle facility projects). For these projects, some general comments can be given:

- The reactors represent a wide selection of types such as PWR, BWR, PHWR, gas cooled/D_2O moderated, water cooled/D_2O moderated, GCR, AGR, VVER, sodium cooled fast reactors and HTGR's both with block type and pebble bed fuel design. The list of reactor projects formerly also included the decommissioning of a plant with two Russian submarine reactors.
- Of the 26 reactor projects, 7 have been completed, i.e. decommissioned to Stage 3 or placed in a "dormancy" status (Stage 2 or Stage 1). Stage 3 implies that the sites have been returned to "green field conditions" or decontaminated completely so as to have been removed from regulatory control. The completed projects continue to be considered as being part of the Programme, as the information arising and the experience from these projects are in the Programme archives. The dormant plants can also continue to generate information and experiences on building/plant degradation and long-term surveillance.
- The fuel facility projects cover 8 reprocessing plants, 5 fuel material plants, 1 fuel storage bay and 2 isotope handling facilities.
- Twenty-seven of the 42 plants in the Programme are to be decommissioned, or are already decommissioned, to Stage 3, namely total dismantling and decontamination.
- Many of the earlier projects in the Programme had to do with experimental or prototype plants. The projects, which have joined the Programme at a later date, were, for understandable reasons, related to plants of a more standardised and commercial character. Even so, there are still significant differences that can be seen in the planning and execution of decommissioning projects. Apart from the differences that can be expected due to the variation in type of plant, the organisational, economic, regulatory and other circumstances prevailing at each site can strongly influence the decommissioning projects.

The main data and characteristics of the participating projects are described in more detail in Appendix 1.

Table 2. **Reactor projects participating in the information exchange of the CPD**

Facility	Type	Operation	Decommissioning stage
1. BR-3 – Mol, Belgium	PWR	1962-1987	Stage 3 (Partial)
2. Gentilly 1 – Canada	Heavy water moderated boiling light water cooled	1967-1982	Variant of Stage 1
3. NPD – Canada	PHWR CANDU	1967-1987	Variant of Stage 1
4. Bugey 1 – France	Gas graphite reactor	1972-1994	Stage 3
5. EL4 – France	Gas-cooled/heavy water moderated	1966-1985	Stage 2
6. G2/G3 Marcoule, France	GCR	1958-1980	Stage 2
7. Melusine – France	Pond research reactor	1988-1993	Stage 3
8. Rapsodie Cadarache, France	Experimental sodium cooled fast breeder reactor	1967-1982	Stage 2
9. KKN Niederaichbach, Germany	Gas cooled/heavy water moderated.	1972-1974	Stage 3
10. MZFR – Karlsruhe, Germany	PFR. Heavy water cooled and moderated.	1965-1984	Stage 3
11. Greifswald, Decommissioning Project – Germany	VVER	1973-1990	Stage 3
12. AVR – Germany	Pebble bed HTGR	1967-1988	Stage 3
13. KNK – Karlsruhe, Germany	Fast breeder reactor	1971-1991	Stage 3
14. HDR – Germany	BWR, nuclear superheat	1969-1971	Stage 3
15. Garigliano – Italy	BWR (Dual cycle)	1964-1978	Stage 3 planned by 2020
16. Latina Italy	GCR (Magnox)	1963-1986	Stage 3 planned by 2020
17. JPDR – Tokai, Japan	BWR	1963-1976	Stage 3
18. Fugen Japan	Light water cooled… Heavy water reactor	1979-2003	Stage 3
19. Tokai 1 – Japan	GCR	1966-1998	Stage 3
20. KRR 1 and 2 Korea	Pool type research reactors	1962-1995 1972-1995	
21. Bohunice A1 – Slovak Rep.	Gas cooled, heavy water moderated.	1972-1979	Stage 1
22. Vandellós 1 – Spain	GCR	1972-1989	Stage 2
23. JEN-1, PIMIC Madrid, Spain	MTR reactor	1958-1984	
24. Taiwan Research Reactor, Chinese Tapei	Light water cooled Heavy water moderated	1973-1988	Partial dismantling
25. WAGR, Sellafield, UK	AGR	1962-1981	Stage 3
26. Prototype Fast Reactor PFR, Dounreay, UK	Sodium cooled fast breeder reactor.	1974-1994	Stage 1
* KWL, Lingen, Germany	BWR (with superheater)	1969-1971	Stage 1
* Shippingport, USA	PWR.	1957-1982	Stage 3
* EBWR, USA	BWR.	1956-1967	Stage 3
* Fort St-Vrain, USA	HTGR.	1976-1989	Stage 3
* Paldiski, Estonia	Soviet submarine PWR	1966-1985	Stage 1

* Reactor projects not participating under the new agreement.

Table 2. **Reactor projects participating in the information exchange of the CPD**

Power or throughput	Project timescale	Cost estimate	Entry into programme	Remarks
41 MWth	1989-2010	M€150 (2000)	1988	EC Pilot project
250 MWe	1984-1986	MCAD 25 (1986)	1985	In dormancy
25 MWe	1987-1988	MCAD 25	1988	In dormancy
540 MWe	1997-2021		2004	
70 MWe	1989-1999	MFRF 550 (1993)	1993	
250 MWe each	1982-1993	MFRF 150 (1990)	1985	Stage 2 achieved
8 MWth	1999-2006	M€20 (2003)	2004	
20 MWth	1983-1994	MFRF 132 (1989)	1985	In dormancy
106 MWe	1988-1995	MDEM 135	1985	Stage 3 achieved
50 MWe	1994-2005	MDEM 440	1989	
8×440 MWe			1992	
15 MWe			1994	
20 MWe	1991-2003	MDEM 500	1997	
100 MWth	1994-1998	MDEM 50	1993	Stage 3 achieved
160 MWe		M€297 (2000)	1985	
210/160 MWe		M€615 (2000)	1999	
90 MWe	1986-1996	MJPY 22500	1985	1981-1986 R&D. Stage 3 achieved
165 MWe	2003-2023		2000	
166 MWE	2001-2017	M€660 (2004)	2002	Timetable depends on availability of
150 MWe			1992	
500 MWe	1992-2003	MESP 14600		
3 MW	1999-2008		2005	Includes other CIEMAT facilities (PIMIC project).
40 MWt	1998-2002			
100 MWt	1983-1998	MGBP 58	1985	
250 MWe			1997	
520 MWth	1985-1988		1985	No longer in CPD.
72 MWe	1985-1989	MUSD 91 (1990)	1985	No longer in CPD.
100 MWt	1986-1996	MUSD 194	1990	No longer in CPD.
330 MWe	1972-1995	MUSD 174	1993	No longer in CPD.
–	1994-		1997	No longer in CPD.

* Reactor projects not participating under the new agreement.

Table 3. **Fuel facility projects participating in the information exchange of the CPD**

Facility	Type	Operation	Decommissioning stage
1. Eurochemic Reprocessing Plant Dessel, Belgium	Reprocessing of fuel.	1966-1974	Stage 3
2. Building 204 Bays Project Chalk River, Canada	Fuel storage pond.	1947 to date	
3. Tunney's Pasture Facility Ottawa, Canada	Isotope handling facility	1952-1983	Stage 3
4. AT-1 La Hague, France	Pilot reprocessing plant for FBR.	1969-1979	Stage 3
5. Radio Chemistry Lab. Fontenay-aux-Roses, France	Reprocessing R&D.	1961-1995	Stage 3
6. ATUE France	Recovery of enriched uranium.	1965-1996	Stage 3
7. Elan IIB France	Manufacture of ^{137}Cs and ^{90}Sr sources.	1970-1973	Stage 2
8. APM Marcoule, France	Pilot reprocessing plant.	-1997	Stage 3
9. UP1 Marcoule, France	Industrial reprocessing plant.	1958-1997	Stage 2
10. WAK Germany	Prototype reprocessing plant.	1971-1990	Stage 3
11. JRTF Tokai, Japan	Reprocessing test facility.	1968-1970	Stage 3
12. Plutonium Fuel Fabrication Facility, Japan	Fabrication of MOX fuels.	1972-2002 (for ATR) 1972-1988 (ex. FBR)	Stage 3
13. Uranium Conservation Facility, Korea	Conversion of yellow cake to UO_2/UF_4.	1982-1992	Stage 3
14. ACL Project Studsvik AB, Sweden	Pu and enriched fuel research.	1963-1997	Stage 3
15. BNFL 204 Primary Separation Plant Sellafield, UK	Reprocessing facility.	1952-1973	Stage 2
16. BNFL Co-precipitation Plant Sellafield, UK	Production of mixed plutonium and UO_2 fuel.	1969-1976	Stage 3
* West Valley, Demonstration Project, USA	Reprocessing plant for LWR fuel.	1966-1972	Stage 3
* FEMP, USA	Hexafluoride reduction plant.	1954-1956	Stage 3

* Fuel facility projects not participating under the new agreement.

Table 3. **Fuel facility projects participating in the information exchange of the CPD**

Power or throughput	Project timescale	Cost estimate	Entry into CPD	Remarks
300 kg/d	1989-2012 Main process bldg	M€ 179 (2003)	1988	Execution by in-house staff
			1997	
	1990-1994	MCAD 13 (1991)	1990	Stage 3 achieved
2 kg/d	1982-1998	MFRF	1985	EC Pilot Project
	1995-2011		1999	
	2000-2007	M€ 30	2004	
	-2012	M€ 450	2004	
500 t U/a	1998-2028	M€ 2000	2002	Main project in France.
			1993	
	1991-2004	MJPY 8600	1991	
10 t MOX/a 1 t MOX/a	-2020 excluding building.		2004	
100 t U/a	2000-2007		2003	
	1998		1999	
Metal = 500 t/a Oxide = 140 t/a	1990-2010	MGBP 90	1990	
50 kg/d	1986-1990	KGBP 2 245 (1990)	1987	Stage 3 achieved
100 t/a	1982-2024	MUSD 1 400	1986	This project is no longer in the CPD Programme
			1993	

* Fuel facility projects not participating under the new agreement.

3.2 Directions in the development of decommissioning projects

As can be seen from the Tables 2 and 3, the projects in the Co-operative Programme have a wide spectrum of characteristics. The circumstances regarding organisation, regulations, economy etc. vary widely even from project to project in the same participating organisation. Of even grater significance is that the types of plant in the various projects are for the most part very different from each other. However, looking back over the recent years, there are some tendencies in commonality in the organisational approach and in the experiences achieved.

Over the 20 years of experience of the Co-operative Programme on Decommissioning, and in particular through the information exchange and review within the TAG, it has become evident that:

- Decommissioning can and has been done in a safe, cost-effective and environmentally friendly manner.
- Current technologies have demonstrated their effectiveness and robust performance in numerous decommissioning activities.
- Feedback of experience on design, construction and operation is a considerable help for reliable planning, cost evaluation and successful realisation of a decommissioning project.
- The dissemination of best practices and sharing of information in international workshops, conferences and specially within the CPD has proven to be a good basis for an effective cooperation and support to master new challenges on decommissioning projects.
- During decommissioning radiological risks are very small in comparison to non-radiological risks.
- Future challenges will require further international cooperation to establish sustainable regulations and guidance to achieve objectives without being burdensome or overly conservative. A consistent, internationally accepted rationale is necessary for the elaboration of concepts and for the derivation of numerical values on clearance, exemption and authorised releases.
- With decommissioning moving towards being a fully mature industrial process, there is a need for increased dialogue among regulators, implementers and international standards organisations.

Regarding technical challenges and the organisational framework for decommissioning, specific trends have been observed over the last decade:

Dismantling of large components

Large contaminated components, for example heat exchangers, steam generators, large tanks etc. have to be segmented into smaller pieces to fit into waste containers for disposal. The segmenting and packaging processes can be time consuming blocking other work and thus placing the removal of the component in question on the critical path in the project (dismantling) time schedule. Often segmenting *in situ* is technically more difficult due to the lack of free space around such components inside containments that are densely packed with equipment. Moreover, because of (generally) higher ambient doses inside the containment segmenting *in situ* can lead to significant dose uptake by the operators.

Against this background, several projects have chosen to remove such components "in one piece" and to segment and package them in waste containers in separate facilities outside the containment. Naturally, this approach is only possible if such an alternative facility is available. Examples of projects where this approach has been extensively used are the MZFR project where large components are transported to the central Waste Management Department (HDB) at Karlsruhe. Also for the Greifswald Project such components are taken to Intermediate Storage North where Storage Hall No. 7 has been specially equipped for the purpose.

Utilisation of robotics

In the early days in the development of the technologies for the decommissioning of nuclear facilities, it had been considered that robotic methods would be extensively used in the dismantling of radioactive components, especially in the high radiation areas in fuel facilities. Experience within the CPD however showed that industrial robots may have a limited applicability in decommissioning, especially due to the non-repetitive tasks that have to be performed in the unstructured and continuously changing environment that characterises decommissioning work. More emphasis is therefore put on the optimisation of proven, commonly available industrial techniques. These techniques are adapted for use in a nuclear environment, with the required reliability and the required safety, in order to increase the comfort for the operators in comparison to working with manually operated tools, while the overall control by the operator is kept. Based on a good co-operation with the non-nuclear industry, excellent results may be obtained.

Challenges for release of alpha contaminated areas

The activities connected with the release from regulatory control of (suspected) alpha contaminated areas in fuel facility projects, have been, in several cases, much more time consuming and labour intensive than envisaged in the original planning of efforts and time schedule. Apart from the difficulty of measurement at the extremely low levels required by most authorities, other types of difficulties encountered have been the reappearance of activity on walls previously declared "clean" and the seepage of contamination into the cracks caused by penetrations in the concrete of cell walls. In case of the latter sometimes several cm of concrete must be removed before the surface can be declared "clean".

Sequential licensing

Some of the major projects in the Programme (e.g. the MZFR, WAK and the Greifswald Projects) that run over a large number of years are executed on the basis of a number of sub-licenses applied for and granted sequentially over the period of dismantling rather than of a single license issued at the start of the project. Thus MZFR is being decommissioned under 8 sub-licenses and the Greifswald VVER's are being dismantled under an even larger number of sub-licenses and permits. While this approach requires very large efforts on the part of the project for the preparation of documents, meetings with the regulatory authorities, revision of documents etc., a possible advantage is that the project management is forced to analyse in advance and in great detail each stage of the project.

Another project, the B204 Primary Separation Plant, is being executed in nine phases with each phase being planned in detail, funding applied for and granted with the work then executed. As this is a very large project, there is considerable advantage in concentrating the planning and execution of the work in a segment at a time.

Another example is that in France an updated decommissioning licence process is employed:

- A unique authorisation by Ministerial decree is given for the whole of the decommissioning process.
- Some key milestones are identified for safety reviews by the Safety Authorities.
- An internal authorisation process at the operator is used to approve work to be performed between each key milestone.

Industrial re-organisation

The time schedules of several projects have been affected quite considerably by re-organisations of various types:

- Privatisation.
- Company re-organisations.
- New company strategy defining different end points to the decommissioning programme.
- Budgetary difficulties.

3.3 Activities of Task Groups

An important functional area of the Co-operative Programme has been the work in the various task groups, particularly the Task Group on Recycling and Re-use and the Task Group on Decommissioning Costs. The results of the work of the task groups have, at the drafting stage, been discussed and analysed at meetings of the TAG.

3.3.1 Task Group on Decommissioning Costs

In 1989, the Co-operative Programme set up a Task Group on Decommissioning Costs in order to identify reasons for the large variations in reported cost estimates on decommissioning projects. In their report, the Task Group made a proposal for a listing of cost items and cost groups that could be the framework for a standardisation. A renewed survey and study was requested by the LC in 1994 and a new Task Group was created accordingly in 1995. The renewed Task Group produced in co-operation with IAEA and EC a new uniform and complete approach to structuring decommissioning costs, which has been published as an interim technical report [4].

However, an attempt to fill the structure with substantial cost data failed. As a result, it was agreed in April 2002 that the activities of the Task Group on Decommissioning Costs should be terminated for the time being.

3.3.2 Task Group on Recycling and Re-use

In 1992, the Programme set up a Task Group to study the recycling and re-use of redundant material from the decommissioning of nuclear facilities and in particular to provide information and insights into the practicality and usefulness of the criteria being developed for the release (clearance) of such material from regulatory control as seen from the perspective of organisations currently engaged in actual decommissioning operations. A report of the work of the Task Group was published in 1996 [5].

Following the release of this report, the Task Group, and in particular the Technical Secretary continued to collect literature and material on this important subject. The group prepared a survey of the international discussions regarding clearance and exemption of materials from nuclear installation decommissioning operations and on the regulatory treatment of technologically enhanced natural occurring radioactive materials, based on literature up to 2001.

3.3.3 Task Group on Decontamination

In October 1992, the Technical Advisory Group of the Co-operative Programme established a Task Group on Decontamination in order to prepare a state-of-the-art report on decontamination in connection with decommissioning. Based on detailed data from some CPD projects a report [6] has been produced in 1999 surveying applied decommissioning techniques and characterising selected techniques for both decontamination of segmented components and for decontamination of building surfaces. While the report itself is freely available, an appendix with the etailed data gathered has been kept within the CPD, in accordance with the CPD Agreement.

3.3.4 Task Group on Release Measurements

The Task Group on Release Measurements was established in December 1996, after a recommendation by the Task Group on Recycling and Re-use, that a specialist group should study the problems that arise in connection with activity measurements at the extremely low levels required by the existing draft/interim release criteria. The Group approach collected relevant data from CPD projects and produced a report giving an overview on equipment and methods for, and applications of, release measurement, as well as presenting case studies and cost information from specific projects [7].

4. SIGNIFICANCE OF THE PROGRAMME FOR THE PARTICIPANTS

The Co-operative Programme covers a broad range of reactors and fuel facilities. The reactors represent almost all types to be found in both research and power production utilisation of atomic energy. The group of fuel facility projects is also very comprehensive and covers material production to storage facilities to reprocessing. Moreover, the local organisational, economic, regulatory, political and other circumstances can differ very widely, indeed even between different projects in the same country.

The feedback from the experience accumulated by this wide variety of decommissioning projects has been organised through the TAG. Key examples demonstrate that the lessons learnt from the CPD activities by the participants have been helpful for individual projects in making project decisions and in many cases have influenced general project directions.

4.1 Lessons learnt from CPD's decommissioning projects

Dry abrasive blasting

The information exchange about decontamination techniques was a clear support when the dry abrasive blasting installation at one of the projects was selected for the decontamination of metal components as a safe, efficient and cost effective technique with minimum production of secondary waste and which could be installed and used in an industrial way. At the same time, melting of metallic material after decontamination could be identified as a technique to characterise the ingots for unconditional release.

Cutting techniques

Experience at several projects taught us that when cutting techniques are considered, it is necessary to make a global evaluation, including all required details for each technique. Machine or tool parameters may indicate good performance characteristics. However, the required preparatory work, the required work organisation, the secondary waste arising or additional constraints may drastically reduce individual performance rates. As such, a very effective cutting tool may still result in a less efficient cutting technique. It is recommended to make an overall evaluation in every case. The same discussions gave adequate information and ideas for developing the required equipment for future cutting of high-level waste storage tanks.

Industrial robots

Experiences in the Co-operative Programme have indicated that the role of robots will be considerably less than expected in the earlier days.

Instead of concentrating on totally remote/robotic methods, the approach seems to be developing to use long handled tools with shielding or to create a less hostile environment by identifying and removing the high sources of radiation as early as possible. The following examples of such experiences can be mentioned:

- A characteristic of the AT1 project to decommission the pilot plant for reprocessing fast breeder fuel was the remote dismantling machine, ATENA. The machine had a 6 m long articulated arm, with a manipulator MA23, later replaced by the heavier duty RD500. ATENA and its manipulators proved to be expensive to build, as well as complicated and expensive to maintain and service. It was seen that it could not be used as a tool in other projects (as had been foreseen) and so it had to be decommissioned and ended up as radioactive waste.

- In the WAK project, where the German fuel reprocessing plant is being decommissioned, a remote dismantling test facility was built for testing equipment and techniques and for training crews for using it on a full scale (cold) model of one of the major cells. The facility was controlled and monitored through a number of TV cameras, monitors and control consoles and was, understandably, very expensive to build and run. Actual experience in high radiation cells has shown that remote dismantling, in spite of this large investment, requires a great deal more manpower than expected.
- In the BR3 project, two robotic arms (one purchased from Norson Power, United Kingdom and one recently from Cybernetix, France) have been purchased. The training and set up procedures were so complicated and expensive that the first robot never entered the controlled area and hence was never used. The second robot is only partially used in defined operations where there are back up solutions available replacing the robot.

However, there has also been successful use of robots, e.g., in the BNFL B204 Primary Separation Plant, for the removal of stainless steel hulls from a storage silo using a remotely operated loading vehicle. BNFL has also successfully integrated industrial robots in the workshop for size reduction and packing of highly radioactive waste in the same project. In addition, BNFL reported the successful deployment of the CODRO (Contact Deployment Remote Operation) approach, which is less expensive and avoids many of the difficulties encountered in a totally robotic operation.

Removal of concrete biological shielding

Irradiated and other concrete in the biological shielding round reactor vessels has been removed in several projects. Many different methods have been used, e.g.:

- Diamond sawing and coring.
- Abrasive water jet.
- Controlled blasting (“soft” explosives).
- Diamond wire sawing.
- Circular saws.
- Remotely operated BROKK machine.

The results within the various projects have been discussed at several TAG meetings, where comparisons have been made from the point of view of productivity factors, worker safety, etc. Experiences have varied, partly affected by site-specific aspects.

Decontamination of concrete surfaces

Several widely varying methods have been utilised for the decontamination of concrete surfaces in the various projects. Apart from the conventional scabbling and other mechanical descaling tools, projects have developed:

- Concrete shavers for floors and walls.
- Methods for automatic application of such tools for increasing productivity.
- Mini electro-hydraulic hammer units for areas of deep penetration of contamination.
- Microwaves for concrete decontamination machines.
- Laser beams for the removal of contaminated concrete surfaces.

The advantages and disadvantages of these methods have been discussed and compared at several TAG meetings.

Use of ventilated suits

Various fuel facility projects have had different approaches to solving the problems associated with the "discomfort" factor in the use of ventilated suits in alpha contaminated areas. It has been noted in some projects that this factor was more restrictive for productivity than the maximum allowable annual exposure to radiation. The approaches have varied from a broad based development programme on such suits with better breathing and cooling air systems to more detailed planning of the component removal strategy to reduce the need for the use of such suits.

Lessons learnt on release criteria

The discussions about release criteria have brought the Programme in contact with various interested groups, regulators, designers, implementers, etc. Decommissioning implementers feel that a set of international standards based on realistic scenarios that make use of available data from existing practices is needed. They also feel that validation/calibration of the models and calculations used to derive risk-based release levels are needed. This should be based on data from existing practices, so that excessive and costly conservatism can be avoided. Implementers also reasoned that beside radiological health risks other types of health and environmental risks should be considered in developing release levels.

In this connection, it was seen that, in some instances, state-of-the-art instrumentation might not be capable of measuring the very low release levels that may result from very conservative risk-based standards.

Decommissioning cost estimates

It has become clear that there is a large potential for making errors and that difficulties can be encountered in performing quick international comparisons of decommissioning cost estimates. Numbers taken at face value, without regard to their context, are easily misunderstood and misinterpreted. It was seen that this was due, among other things, to the fact that there was no standardised listing of cost items established specifically for decommissioning projects. Such a standardised list would facilitate communication, promote uniformity and avoid inconsistency or contradiction of results or conclusions of cost evaluations for decommissioning projects carried out for specific purposes by different groups.

Public relations and public involvement

The projects have learned that specific activities should be deployed in the areas of public relations and public involvement. Special attention is required, however, with respect to some ethical aspects of a public involvement strategy, which might be translated into a number of principles, i.e. fairness, openness, volunteerism, shared decision-making and commitment to safety.

In general, learning about the difficulties and the "errors" made or the "failures" that occurred in other projects helps to avoid these difficulties and failures in their own project.

4.2 Conclusion

As is demonstrated above, the knowledge and information gleaned from the Programme are both generally applicable and of common interest at one level, and are project specific at another. Even in the case of project specific problems, the administrative approaches and manner in which they are solved are of interest to the other participants.

In addition to the tangible benefits listed above, there is the added value of personal interaction with experienced people from a wide cross-section of the decommissioning community. The TAG meetings promote exchange of information and the relationships built at these meetings enable access to this information on a detailed and personal basis with the necessary confidentiality assured. This is an invaluable asset.

5. FUTURE OF THE CO-OPERATIVE PROGRAMME

Looking back over the twenty years since the OECD Nuclear Energy Agency established the Co-operative Programme on Decommissioning, the CPD has functioned as the main international forum for the exchange of technical and other information arising from nuclear decommissioning projects. By the dismantling and release from regulatory control (Stage 3 decommissioning) of a number of diverse nuclear facilities, the Programme has been able to demonstrate in practice that nuclear decommissioning can be performed safely both for the workers and the public, and that this can be done at reasonable costs in an environmentally friendly fashion.

Decommissioning of nuclear installations will become, in a few decades, one of the important sectors in the nuclear market and decommissioning experience probably will be of equal importance to the design, construction and operation of nuclear installations.

The rise of participation in the Co-operative Programme on Decommissioning from 10 projects in 1985 to 42 projects 20 years later and with several new projects knocking on the door for admission clearly demonstrates the added value participating organisations attach to the unique information exchange the Programme provides.

Some of the reasons for this development obviously are that:

- Participation is purely voluntary, based on the principle of “give and take”.
- The TAG meetings are a unique forum for free, open and frank discussions in depth as well as for cross-comparison of practical shop-floor issues and general approaches.
- The Programme has been able to develop accepted methods of respecting the confidentiality of sensitive information.
- The various functions within the Programme, e.g. the role of the Programme Co-ordinator, the TAG meetings, the Task Groups, have continually evolved to meet the requirements of the changes taking place in the world of nuclear decommissioning, e.g. commercialisation of the nuclear decommissioning industry.

The organisation of the Co-operative Programme as a whole can also be expected to evolve to meet the requirements that may arise. In this frame, participation in the Co-operative Programme is of key importance in order to be acquainted and updated about decommissioning technologies, costs and safety-related aspects. It will help those undertaking decommissioning to make reliable plans and cost evaluation and will help to improve safety. This is particularly important and essential for countries with limited resources for the decommissioning of their nuclear installations.

It is also of great importance for those undertaking decommissioning to belong to a group with wide experience and to follow its discussions; the best solutions can be brought to bear upon the problems experienced.

As for the future, it is foreseen that participation in the Co-operative Programme will continue to grow. With it, the basic information exchange activities will continue in their current form, based on confidentiality, and a "give and take" approach.

New projects are welcome to enter the information exchange. Information on how to join the CPD can be obtained from the NEA Secretariat.

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Appendix 1

DESCRIPTION OF DECOMMISSIONING PROJECTS

The projects participating in the Co-operative Programme as of November 2006, are described in this Appendix, under the following headings:

- A1.1 Reactor Projects in Progress (19 projects).
- A1.2 Completed Reactor Projects, i.e. decommissioned to Stage 3 or placed in a "dormancy" status (Stage 2 or Stage 1) (7 projects).
- A1.3 Fuel Facility Projects in Progress (14 projects).
- A1.4 Completed Fuel Facility Projects (2 projects).

In addition, short descriptions are provided for projects which had been participating but left the Co-operative Programme before November 2006:

- A1.5 Former CPD projects which are no longer participating in the Programme (7 projects).

A1.1Reactor projects in progress

A1.1.1 BR3 PWR Reactor, Belgium

The BR3 reactor was the first PWR installed and operated in Europe. It is a low rated plant (40 MWth, 10.5 MWe net), but presenting all the features of a commercial power plant of the pressurised water type. The reactor was used at the beginning of its lifetime as training facility for future NPP operators. Later on, it was also used as test bench, in full PWR conditions, for new types of nuclear fuel (e.g. MOX, consumable poison, high burn-up, etc.).

The reactor was shutdown in 1987 after 25 years of operation. In 1989, the BR3 was selected by the European Commission as a pilot decommissioning project in the framework of its 5-year plan on decommissioning of nuclear installations. The pilot decommissioning project started in 90-91 by a pre-dismantling decontamination of the primary loop. Afterwards, the dismantling of a first reactor internal, the thermal shield, presenting high radioactivity and dose rate, was studied and then carried out. This dismantling, i.e. cutting the piece into parts fitting in the final radioactive waste package (400 l drums), was done remotely and under water. The water was used as shielding for the operators against the radiation coming from the piece. The dismantling was successfully completed in 1991, using three different techniques: the plasma arc torch, electric discharge machining (or sparking erosion) and the mechanical milling.

These three techniques were compared for what concerns the generated secondary waste, the dose uptake, the manpower and the overall costs. This comparison led to prefer mechanical cutting for the subsequent operations.

Afterwards, the remaining internals, some of them presenting even higher radioactivity (up to 4 Ci/kg or 150 GBq/kg ^{60}Co) due to their closer proximity to the reactor core, were dismantled using remote controlled band sawing and circular sawing. The BR3 disposing of a second set of reactor internals, for historical and experimental reasons, the dismantling of these internals took place directly after the preceding operation. This dismantling was also part of a contract with the European Commission, as follow-up of its Research and Technological Development Programme. All the internal pieces were packaged in 400 litre drums, then conditioned by grouting with cement and stored at the Belgian intermediate storage facility for radioactive waste (Belgoprocess). The dismantling of the Reactor Pressure Vessel (RPV) was completed in the middle of 2000. After removal as one piece into the refuelling channel, it was remotely segmented under water, using mainly mechanical techniques. The produced pieces were packaged in 400 l drums.

The remaining large dismantling activities concern the dismantling of the RPV top cover, the RPV bottom, the big components of the primary loops (Steam Generator, Pressuriser, pumps housings) and the RPV surrounding Neutron Shield Tank (NST).

Remote controlled tools will be used, the technical choice being a High Pressure Water Jet cutting tool deployed by a Maestro teleoperated arm.

In parallel with this pilot project, the dismantling of contaminated loops and equipment has been going on.

For the decontamination of stainless steel and carbon steel pieces an industrial chemical decontamination unit called MEDOC has been put in operation in 1999. The MEDOC process (for Metal Decontamination by Oxidation using Cerium) is based on the use of Cerium IV as strong oxidant in sulphuric acid with continuous regeneration by ozone. Up to now, about 80% of the treated mass could be unconditionally cleared and sold to a scrap dealer; the remaining 20% have a residual radioactivity lower than 1 Bq/g and may by cleared after melting in a nuclear foundry.

In addition the process has been modified to allow decontamination of carbon steel. At the end of 2001, the MEDOC process was used to further decontaminate the plant steam generator in one piece.

Full-scale tests were also carried out for the dismantling of concrete, representing the major part - in mass- of nuclear facilities. The dismantling of activated concrete, i.e. able to present a quite significant dose rate or high contamination hazard, was tested by using different methods, from the controlled blasting to the remote operated jackhammer and excavator. However, most of the concrete is surface contaminated concrete (with low levels of contamination) where the major risk is the spread of contamination, by the distribution of fine dust possibly contaminated.

A1.1.2 EL- 4 Brennilis, France

The Brennilis Plant was a 73 MWe heavy water reactor that operated between 1967 and 1985. As part of the EDF Decommissioning Programme, this reactor will be decommissioned to Stage 3.

The original programme was to decommission the facility to Stage 2 i.e. the non nuclear buildings (offices etc.) would be demolished along with the nuclear classified buildings (effluent treatment station ETS, spent fuel building SFB, the solid waste store SWS and stack) and cleaned to set target levels and then demolished. It was intended to achieve Stage 2 status by the end of 2005.

Three of the nuclear buildings have been decontaminated. These are the ETS, the SWS and the SFB. The SWS was demolished in 2002 after total declassification. The ETS was demolished in 2004. The SFB is planned to be demolished by February 2005. The decontamination of the basement of ETS and the deep pits in the SFB remain to be completed by mid 2005.

With regard to the change in strategy to a Stage 3 programme, its feasibility has been confirmed and funding secured. Technical specifications for the required work will be prepared in 2005 with a call for tenders in 2006 with a completion date of 2018 for Stage 3 decommissioning.

The Stage 2 licence has been modified giving an additional 3 years to achieve Stage 2 and to partly complete some preparatory work for Stage 3. A licence application has been submitted to the regulatory authorities for the Stage 3 green field status. This is expected to be granted in early 2006.

During the achievement of Stage 2 from 10/1997 to 07/2003:

- Some 400 t of L or ML waste and 1200 t of VLLW was produced. The rates at which dismantling produced the wastes was 130 t per hour when no special tools were used and 200 t per hour when special equipment such as scaffolding and handling tools were used.
- Four categories were used to define the category of concrete decontamination required:
  - Category 0: Surface with no radioactive contamination.
Dust removal (2 300 m^2).
  - Category 1: Surface with suspected radioactive contamination.
Removal of 2 mm of concrete (11 000 m^2).
  - Category 2: Surface with suspected liquid superficial radioactive contamination.
Removal of a maximum of 6 mm of concrete (4 300 m^2).
  - Category 3: Surface with radioactive contamination, possibly deep.
Removal of over 6 mm of concrete (5 000 m^2).

The quantity of waste produced during concrete decontamination was:

- 150 t of M or LLW.
- 2 500 t of VLLW.
- 540 t of concrete blocks (VLLW).
- Some 730 000 hours were worked with up to 150 workers employed during one year (average workforce 60). The total dose received was 115.5 mSv, 0.3 mSv/year/worker. Three quarters of the dose was received during dismantling operations.

A1.1.3 Bugey 1, France

Bugey 1 was a 540 MWe gas graphite reactor that operated from 1972 to 1994. It is currently in a Stage1 (practically Stage 2) status. Some of the technical characteristics of the plant are:

- The reactor is in a pre-stressed concrete pressure vessel with internal dimensions 17 m diameter, 40 m high and external dimensions 28 m diameter and 56 m high.
- The reactor pressure vessel contains the core of 2 600 t of graphite (15 m diameter, 9 m high) above the heat exchangers that are integrated into the vessel.

The vessel internals include 942 steel guide tubes above the core, the “corset” around the graphite core, the 15 780 hexagonal graphite bricks making up the core, the hot plenum of the gases exiting the core as well as the heat exchangers on their metallic cylindrical support.

The current status is as follows:

- Fuel has been removed.
- Control rods are still in place.
- High activity waste is temporarily stored in the upper slab cavities.
- Coolant turbines, power generating and auxiliary equipment have been removed.
- The pump station has been demolished (2003-2004).

Basic design studies have been performed by the newly established division of EDF to manage decommissioning, the *Centre d’ingénierie, déconstruction et environnement* (CIDEN). These studies, based on proposals from experienced nuclear decommissioning contractors, indicate:

- A first phase to dismantle the reactor vessel internals, a second phase to clean up the buildings.
- Remote dismantling of the core and associated parts.
- Manual dismantling of the heat exchangers etc. in the lower part of the vessel.
- Removal of activated concrete before the non-activated.

A basic design review meeting held in early 2003 concluded that the “open” scenario is the reference scenario. This scenario consists of dismantling the internals using an opening to be cut out in the upper reactor slab.

The conceptual design will begin in the middle of 2005. In preparation of this conceptual design additional pressure vessel radiological measures have been performed (from 2003 to late 2004):

- Metallic samples have been taken from the bottom of the reactor vessel.
- 10 cores have been taken in the reinforced concrete using dry drilling. Out of these 10 cores, 6 reached the inside of the vessel allowing samples to be taken of the metallic metal liner.

The concrete and metal samples have been sent to CEA-SERMA for analysis.

The specification for the final dismantling of the electromechanical equipment and the removal of operational waste is underway with contracts to be let in late 2005 and mid 2006.

A decommissioning licence is expected in early 2007. Preparatory works on the decommissioning infrastructure on site will take place during 2005-2007. The reactor internals will be dismantled during 2008-2015. The active concrete will be removed between 2015 and 2018, while concrete structural demolition and site restoration will be carried out from 2019 to 2021.

A1.1.4 Melusine, France

This is a multi-pupose pond research reactor. This reactor, commissioned in 1959 (1 MWth) had its power increased to 8 MWth in 1971 (length 15 m, height 9 m, pond walls thickness 0.8 m). It was dedicated to fundamental research, technological and materials irradiations and production of radioisotopes.

Due to demography evolution and expansion of Grenoble city, the CEA’s Grenoble Centre is now inside the town. The CEA decided in 1995 to stop all nuclear activities by 2015 and was faced to the denuclearisation and the decommissioning of all activities (research reactors, hot cells, research laboratories, effluent and waste treatment station).

After operations starting in 1988 and ending in 1993, the fuel was removed, some experimental devices were treated and circuits rinsed. The facility was then kept under surveillance from 1993 to 1999 due to budget constraints.

In the facility, some experimental devices were accumulated during its lifetime and were not treated during shutdown operations (they were stored all around the pond), and the remaining cut materials were lying on the pond bottom. During the dormancy period, it was necessary to refurbish the water circuit and to put the facility into a safe state in accordance with industrial requirements (fire detection, electricity etc.). In 2000, it was decided to restart decommissioning to achieve Stage 3.

The first step was to clean and empty the pond and to get from the Safety Authorities the decree allowing the CEA to decommission the facility. The pond was cleaned and emptied of its remaining materials and water by February 2003.

The decommissioning decree was obtained in January 2004. The removal of the ceramic tiles covering the concrete reactor walls and floor started in June 2004.

Core boring to extract the canal nozzles is ongoing with difficulties being experienced coming from the cutting of the pre-tensioned cables (poor concrete injection of the assembly protective sleeve/ pre-tensioned cable) causing two months delay.

The final cleanup of the reactor pond and associated cells started in 2005 with completion having to be achieved by 2006 for the delicensing (declassification of the facility). Then the facility can be demolished using conventional methods.

Wastes forecast Phases 2 and 3		
Wastes	**Container types**	**Numbers**
FA	200 l inc	22
FA	120 l PEHD	120
TFA	200 l comp	6
TFA	2 m^3	80
FA	5 m^3	10
FA	Bonbonne 30 l	10
HA/MA	Caisson prébétonné	1
TFA	Casier	12
FA	10 m^3	1
VLLW (TFA)	Big-bag 1 l	80

Dosimetry forecast	
Phase 2	29.2 man-mSv
Phase 3	2.4 man-mSv
Total	31.5 man-mSv

Up to mid 2004:

- The wastes produced at the different phases were sent to the waste treatment station STED for treatment and disposal:
  - Phase 1 MLW.
  - Phase 2 LLW and VLLW.
  - Phase 3 VLLW and conventional.
- The amounts of waste types produced were:
  - HLW 0.2 m^3 CSA, 37-46 l bins.
  - MLW 14 m^3 CSA.
  - LLW 74 m^3 CSA and Centraco (burnable).
  - VLLW 194 m^3 CSTFA.
  - Conventional waste 1 500 m^3 after declassification.
  - Liquid waste 200 m^3 after control, doubtful sewer and discharge after control to the Isere river. Annual limits for discharge: H^3: 2 GBq.
    Beta gamma emitters: 0.05 GBq.
    Alpha: 5×10^{-3} GBq.

The total cost of the project is estimated at M€20 (2003).

A1.1.5 MZFR, Germany

The MZFR was a 200 MWth (50 MWe) pressurised heavy water reactor that operated at the Kernforschungszentrum Karlsruhe from 1966 to 1984. The plant is being decommissioned to a Stage 3 status under a series of eight sub-licences. Work under the first six sub-licences has been completed. Currently, the reactor vessel and internals are being dismantled under sub-licence seven.

During the earlier work, all D_2O had been removed, the systems dried, the cooling towers demolished and the water treatment plant dismantled. The turbine hall was cleared and handed over to the WAK project for the installation of a test facility.

Under the fourth sub-licence, the secondary circuit and the auxiliary systems were dismantled and equipment in the pool building was removed. A major task was the chemical decontamination of the primary system, where an average decontamination factor of 20 was obtained. Another important activity under the same sub-licence was the dismantling of the 4 D_2O enrichment columns, each 12 m in height and 1.5 m in diameter.

The safeguard requirements at the site have been removed and the security fence taken down.

The criteria for release of components from the plant are that the surface activity should be less than 0.5 Bq/cm^2. This is applied when all surfaces are accessible for measurement. Complex geometry components are sent to a special measurement unit. Swipe tests are taken on “non-contaminated” systems in the controlled area.

A steel caisson with a transfer lock has been built to allow airtight docking at the reactor building. This is large enough to allow 20-foot containers, weighing up to 15 t, to be transferred in and out of the building. A larger transfer lock is being built, to allow the transfer of even larger items, e.g. steam generators, which will be removed in one piece and sent to the Waste Treatment Department for segmenting. After the removal of the steam generators, a packing station will be established with its own ventilation system.

The 6th licence covered mainly the dismantling of the primary system and all auxiliary systems inside the reactor building. It also covered other items, such as the dismantling of auxiliary systems in the auxiliary building, installation of the large materials lock and a new entry area on the reactor building, etc. During this work, the two 55 tons steam generators, the 20 t pressuriser and the two main coolant pumps were removed. In the scope of the sixth decommissioning step, the decontamination measures will be checked by release measurements by independent experts. After that, the parts of the auxiliary buildings, which are declared clean, can be released from the controlled zone.

The on-going dismantling of the reactor vessel internals under sub-licence seven, is complicated by the fact that there are 121 vertical coolant channels for fuel assemblies (without fuel) and 18 control rod absorbers in 18 guide tubes that are at a 20° angle to the vertical. Outside the vessel head extends a drive and position indication tube for each absorber.

The following procedure was therefore adopted:

- The drive and position indication tubes were removed first. The lower ends of the tubes being irradiated, these ends were cut off and packed into waste drums.
- Each fuel assembly was lifted into a shielded bell, which is fitted with hoists and suitable grabs as well as a sliding gate at the bottom.
- The shielded bell is moved to the transfer site, where it is aligned to a shielded tube in a tilting device. The fuel assembly is lowered into a transport cartridge in the shielded tube. The cartridge is closed with a lid that is seal welded for preventing spread of contamination. The shielded tube is turned from the vertical to a horizontal position, aligned with a transport device. The cartridge with the fuel element is transferred into the transport device for transport to the waste treatment department (HDB) of Karlsruhe.
- The coolant channels, absorbers and absorber guide tubes will be removed later.

The removal of the rod shaped components started in April 2000. The equipment for dry dismantling of the RPV-components (Lid, upper spacer, lower spacer, RPV) was installed. The band saw was tested in the installed state at MZFR and started with cutting of the RPV-lid late in 2001.

The equipment for wet dismantling (cutting and water) of the moderator tank and the thermal shield has been manufactured. The plasma-cutting device was tested at the University of Hannover. The equipment for plasma cutting was tested in VAK during May 2001.

The removal of the RPV and its internals was completed in 2003. After that, the measures of the eighth step started: the dismantling of the activated biological shield, removal of the infrastructures, decontamination and release measurement of all surfaces and the demolition of all buildings as well as the clean up of the site.

A1.1.6 Greifswald and Rheinsberg, Germany

There are eight 440 MWe pressurised-water reactors of the Russian WWER type at Greifswald, and one 70 MWe at Rheinsberg. They were shut down after the reunification of Germany in 1990, mainly due to a lack of political acceptance and secured financing for refurbishment. Energiewerke Nord GmbH was created to decommission these plants in a socially acceptable form.

Four of the eight Greifswald reactors had been in operation between 1973-1990. The fifth, which was of a more recent design, had been started in 1989. Unit 6 was ready for operation, while the other two were in the process of construction. The Rheinsberg WWER had been in operation since 1966. The plants have leak-tight enclosures which, however, are not comparable with “containments” as on plants in the West.

Direct dismantling was chosen because of lower costs, lower dose commitment and lower volumes of radioactive waste than for the alternative of safe enclosure and deferred dismantling. This is mainly due to the design and site-specific conditions. Direct dismantling has also advantages from the point of view of continued employment for the work force. The project itself can be divided into three phases, each phase conducted under a number of sub-licences:

- The *post-operation phase* comprises: operation of all systems relevant to the safe storage of fuel elements, the removal of fuel elements, conditioning of operational waste, dismantling of not relevant systems (mainly inactive) and system decontamination.
- The *dismantling phase* comprises: the dismantling of the contaminated systems, the remote dismantling and conditioning of dismantled material.
- The *site restoration phase* comprises: dismantling of remaining systems, building decontamination and demolition and finally the restoration or adaptation of the site for other uses.

Currently, the project is in the dismantling phase. Some of the main project activities are described below:

- A central feature of the decommissioning strategy at Greifswald is the Interim Storage Facility North (ISN), which allows the cutting out of large components from the systems for interim storage at ISN, where they can later be treated, when convenient. The ISN will also house fuel elements. The ISN has been in operation since 1998. The operation had to be suspended briefly due to the lack of a licence (under Article 37) from the European Commission. In September 1999, the operation was restarted. The ISN has 8 halls, each with a storage capacity of 25 000 m^3.
- Under the current licence, all fuel elements were transferred out of the wet on-site storage into dry storage in CASTOR casks by the end of 2004. In order to begin dismantling works under easier safety restrictions, all fuel elements at the various reactor unites were transferred into the wet storage first, and are thereafter being put in CASTOR casks, as they become available.
- Decontamination is a much-practised approach in the project. Even during operation, full system decontamination had been utilised on all units. All loops were decontaminated and hot spots removed before dismantling. The electrochemical decontamination was completed

in 1999, with an average DF of 9 on the steam generators, 4 on the main cooling pumps and a reduction of dose rate from the primary system pipes by 35%.

- Remote dismantling will be used on the reactors and the internals of Units 1-4 of the Greifswald plant. Prior to this the equipment to be used is tested in Unit 5, which had only been in operation a short while.
- To date a total of 21 000 t of material has been dismantled at Greifswald. Of this about 5 400 t can be released without restrictions, another 5 600 t is "suspected" material, while about 10 000 t is contaminated waste. No activated material has yet been dismantled. The total dismantled material at Rheinsberg was about 10 500 t.
- There has been a drastic reduction in the work force at Greifswald over the years (from 5 000 to 1 200 workers). This has been achieved by retirement schemes, privatisation of technical and service work, and some unavoidable dismissals.

After initial difficulties due to the unplanned shut-down, the project has proceeded very well; all major licenses have been obtained, the material treatment with especially the release of material is working very well, the treatment of operational waste and loading of spent fuel for dry storage is going on and different site re-use projects are proceeding.

A1.1.7 Arbeitsgemeinschaft Versuchsreaktor (AVR), Germany

The 15 MWe AVR of Arbeitsgemeinschaft Versuchsreaktor in the direct neighbourhood of the Julich Research Centre (FZJ) was a high-temperature, helium-cooled reactor with spherical fuel (pebble bed) developed in Germany. This experimental reactor operated between 1967 and 1988. A licence application has been made for placing the plant in a Stage-1 status. The licence to transform the plant to safe store conditions (Stage 1-2) was granted in March 1994 and comprises two different phases:

- De-fuelling the reactor and inspecting for residual fuel, in parallel dismantling the secondary and auxiliary systems outside of the reactor building.
- Dismantling the auxiliary systems inside the reactor building and closing the reactor vessel.

In contrast to most other decommissioning projects, de-fuelling of the AVR was part of its decommissioning licence and not included in the operational phase. The AVR fuel consisted of approximately 100 000 "pebbles". The de-fuelling operation had been planned to take 19 months, but actually took 4 years to execute. This operation was followed by a reactor cavity inspection, before which a radial hole was bored into the core cavity to facilitate inspection. Technical problems caused many delays.

The inspection revealed that the core bottom was cracked and many pebbles had sunk down and were trapped in the cracks. Some pebbles could be removed but most remain. AVR got the licence to leave them in place for safe store. Unforeseen activities had to be undertaken in order to remove stuck pebbles and a large amount of pebble dust from the fuel discharge line and other parts of the fuel handling system, causing a further extension of the time needed to complete the residual fuel inspection of the reactor.

Traces of strontium and other fission products were found in rainwater in the ground and also in the gap between the reactor building and the hot workshop. This contamination probably took place during the steam generator repair of 1978. This also delayed the fuel inspection.

Project activities had to be reduced to a minimum during the latter half of 1999 because FZJ, then still included in the budgeting of AVR, reduced the decommissioning project personnel budget to about 60% of the agreed level, on instructions from the Federal Ministry of Research. After legal action, the budget was restored to its original level. Again, the fuel inspection was delayed.

All in all, the inspection of the residual fuel took 2 years and 2 months to be completed rather than the planned 6 months.

A consequence of the above affairs was that the Nordrhein-Westfalen Land government declared their preparedness to increase their share of the project financing from 10% to 30% for a total Stage 3 decommissioning of the plant. For this, there was then increased support.

The 15 small companies that were the stakeholders in the AVR had always wanted to proceed in this direction and at the same time intended to terminate their engagement in the project. Eventually in May 2003, the AVR company was taken over by Energiewerke Nord (EWN) company of Greifswald (see above). The decommissioning goal was definitely changed to Stage 3 with the new strategy to remove the RPV including all internals in one piece and to place it intermediately into an interim storage facility that has yet to be constructed at either the neighbouring FZJ site or on site.

The provision of the new facilities at AVR to serve this need have already been licensed as a supplement under the existing safe store licence since this will also facilitate the safe store decommissioning tasks still to be completed. The facilities comprise an airlock building next to and extending over the top of the reactor building, the creation of an upward transportation route out of the containment and new improved ventilation and air exhaust facilities.

In view of new studies undertaken concerning the safety of AVR in the case of civil aircraft attacks and earthquakes, the supplementary licence requires the grouting of the RPV with low density concrete. This will immobilise the fine ^{90}Sr laiden graphite dust inside the vessel and will thus also be an important feature when lifting the RPV.

The construction of the new facilities has begun while the dismantling of auxiliary systems inside the reactor building continues and a new Stage 3 licence is being filed. It is expected that green field conditions will be restored on site before 2015.

A1.1.8 KNK, Germany

The KNK plant was “compact” sodium cooled nuclear reactor, used to develop sodium technology first with a thermal core and later with fast breeder fuel elements. It operated between 1971 and 1991. It is being decommissioned to a Stage 3 status (green fields), under a series of sequential sub-licences. All fuel (both used and new) was removed from the site under the operating licence. Other core internals, like the absorber and reflector elements, core support plate inserts, etc., were also taken out under the same licence.

The reactor plant has a primary system (in the reactor building), a secondary system with steam generators (in the steam generator building) and a tertiary system in the turbine hall.

The first four sub-licences mainly covered:

- Dismantling of the conventional part of the plant (tertiary system).
- Removal of the fence (in common with the MZFR).
- Construction and operation of a plant for discharging the secondary sodium. 50 t of secondary sodium have been removed for disposal in 200 l drums.
- Discharge of the primary sodium for disposal.
- Demolition of the stack.
- Dismantling of the fuel-handling machine.

The fifth and sixth sub-licences comprised the disassembly and disposal of the secondary systems, N_2 cooling system, the turbine and steam generator halls, etc. The seventh sub-licence was to prepare the dismantling of the primary systems, while the actual dismantling of those systems is taking place under the eighth sub-licence. Ongoing work is concentrated to the reactor building and the dismantling of the primary cell and sodium-cleaning cell has been completed. Dismantling of the reactor internals is ongoing (mid 2006). Remaining quantities of sodium call for special attention e.g. washing under inert (Nitrogen) atmosphere and the use of cold cutting techniques.

Sub-licence nine covers the dismantling of the reactor vessel, which is a double walled vessel. Finally, the high density concrete Bioshield will be demolished in the tenth sub-licence, which will also cover the decontamination and demolition of the remaining buildings to achieve a green field status by the end of the year 2010.

KNK is owned by FZK but was operated by the company KBG, which was a subsidiary of the local utilities. Meanwhile, the services of the operational company KBG has been terminated. All further dismantling activities are done under the leadership of the Forschungszentrum Karlsruhe.

A1.1.9 Garigliano, Italy

Garigliano Power Plant has a 160 MWe, dual-cycle boiling-water reactor that was taken into operation in April 1964. The nuclear section, consisting of the reactor, the two steam generators and the nuclear auxiliary systems, is contained in a 49-m diameter spherical secondary containment. The reactor was shut down in 1978; in 1982, it was decided to place the plant in safe storage, SAFSTOR.

At the Garigliano site:

- Spent fuel has been shipped off site.
- Safe enclosure of reactor and turbine buildings was reached in 1998. The reactor building containment has been isolated from the other buildings and ventilated by utilising the temperature and pressure differences between the inside and outside of the containment.
- Radioactive waste treated at site consists of:
  - LLW: 796 packages (320 litres) of super-compacted Dry Active Waste (DAWs), previously stored in 2 429 drums (220 litres), 86 packages (320 litres) of non-compressible DAWs, after sorting and monitoring.
  - ILW: 280 m^3 of sludges, concentrates and resins, retrieved from the storage tanks and cemented in a MOWA plant, resulting in: 399 shielded packages of conditioned sludge, 255 unshielded packages of conditioned evaporated bottoms, 767 shielded packages of conditioned resins, The shields are removable before final disposal of the packages, The treatment took 60 weeks of operation of the MOWA plant. The total cost of treatment of the ILW is about 5×10^9 ITL (approximately 2,5 MEuro).
  - HLW: 4 t of highly activated materials (fuel channels, control rods, in-core parts, etc.) were retrieved from the storage trench and cemented in 6 concrete containers (50 t, 15 m^3 each).
- Other LLW waste is stored at plant site.
  - 600 t of material has been released from the controlled area at a release limit of 1 Bq/g or cm^2 (β/γ) or 0.1 Bq/g or cm^2 (α).
  - Chemical and mechanical decontamination methods have been tested for stainless and carbon steel large tanks.

Plans had also been detailed for the actions to be taken to reach the passive safe enclosure condition in the year 2003. The SAFSTOR strategy had to be abandoned, when SOGIN (the company created by the Italian Government for the post shutdown management of power reactors) received the following new directives from the government in December 1999:

- The decommissioning strategy is changed from SAFSTOR to DECON, with the target to release all the Italian nuclear sites, free of radiological constraints, by the year 2020.
- Operational waste has to be treated and conditioned within the year 2009, for disposal in the national repository.
- The additional costs originated by the acceleration of the decommissioning plans will be compensated by a levy on the energy price, which is established and controlled by the Italian Authority for the Energy Sector (Decree issued in January 26, 2000).

The December 1999 governmental guidelines foresaw a national LLW repository, together with an Interim Storage for spent fuel and HLW, available in Italy starting from January 2009. The change of strategy required the definition of new decommissioning plans, adapting the speed of the process to the milestones established by the governmental plan and specifically:

- Start of construction of the national repository: June 2005.
- Operation of the national repository: January 2009.
- Final release of sites: within the year 2020.

Considering the change of decommissioning strategy, the licensing process has to be restarted by submitting a new application for the one-phase decommissioning authorisation. The new application has been submitted by July 2001 for the Garigliano decommissioning project, with the aim to obtain the permits related to the one-phase decommissioning strategy within the year 2003. For Garigliano, the operating licence to bring the plant in the safe enclosure conditions is not valid any more; the clearance levels contained in this licence cannot be used any more for the release of solid materials.

The logic of decommissioning planning will have to be flexible in order to accommodate significant delays or a change in the strategy, which may imply a significant problem of waste management.

The outstanding priorities are:

- Inventory and characterisation of plant contamination.
- Waste management.
- Waste characterisation.
- Dismantling and decontamination technologies.
- Technology for release measurement.
- Criteria for final site release.

The change of decommissioning strategy enhanced the need to clarify some critical issues for decommissioning, due to their impact on decommissioning plans:

- Regulation for the management of radioactive waste and dismantled materials.
- Clearance criteria for the release of solid materials and criteria for final site release.
- National repository acceptance criteria for disposal of waste and dismantled materials.

A1.1.10 Latina, Italy

Latina was a 210 MWe* GCR that operated between 1963 and 1986. Its definite closure was decided by the Italian Government in 1990. It had been planned to achieve a safe enclosure status by the year 2004. The safe enclosure period was expected to be 40 years, after which the plant would have been dismantled and the site released.

At the Latina site:

- Spent fuel has been shipped off site.
- Some preliminary decommissioning activities began in 1992 and concerned the dismantling of some systems and components no longer safety-related, such as:
  - Water-steam piping and auxiliary piping.
  - Thermal insulation of boilers and primary circuit ducts.
  - Biological shield fans.
  - Fuel charge/discharge machines.
  - CO_2 production and storage plant.

The Italian Control Authority (ANPA) also authorised activities addressed to demonstrate the feasibility of some operations, and to test the adequacy of the operational procedures. These activities are:

- Dismantling of two by-pass ducts of the primary circuit.
- Decontamination of two sections of the spent fuel pool.

These two activities have been performed with positive results, so that useful experience has been acquired in the implementation of decontamination techniques of concrete structures and steel components, as well as in the field of plasma cutting.

- Radioactive waste treated at site consists of 500 packages (380 litres) of super-compacted DAWs (LLW), previously stored in 1 512 drums (220 litres).
- Other untreated radioactive waste (mainly sludge and Magnox debris) is stored at plant site.

Plans had also been detailed for the actions to be taken to bring the plant in the safe enclosure condition in the year 2006.

At Latina, decommissioning activities had not proceeded to the same extent as at Garigliano. The detailed design for dismantling the primary gas ducts had been approved by the authorities provided an environment impact assessment was made before start of dismantling.

As has been the case with the Garigliano decommissioning project, Latina is also affected by the directives of the Italian Government in December 1999 and January 2000. In the case of Latina, the new application for decommissioning to site release is planned to be submitted by December 2001.

Otherwise the text above describing the situation at Garigliano is fully applicable to Latina.

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* In 1971, thermal power has been reduced in order to reduce reactor temperature to prevent corrosion of the core supporting structure. The gross capacity was then reduced to 160 MWe.

A1.1.11 Fugen, Japan

The Fugen (called on Advanced Thermal Reactor (ATR)) is a 165 MWe, heavy water moderated, light water cooled, pressure tube type reactor, owned by the Japan Nuclear Cycle Development Institute (JNC). It has been in commercial operation since 1979. A major characteristic of its operation has been the use of MOX fuel, including some containing plutonium from Fugen spent fuel. It has operated quite successfully (with a 62% average load factor), but a governmental decision was taken in 1998 to stop further work on the ATR. So the Fugen was shut down in 2003 at the latest. As a basic preparatory step, the planning of its decommissioning has started.

The current activities are in the areas of:

- Evaluation of radioactive inventory.
- Study and planning of dismantling.
- Waste management of decommissioning waste.
- Setting up an engineering support system.

The irradiated inventory is being estimated by well-known calculation codes as well as by the measurement of flux by irradiating foils. The contamination inventory is based on sampling (of concrete) and gamma measurements on equipment and systems. Based on these calculations and measurements, it is estimated that the dismantling the Fugen reactor will produce 370 000 t of waste, of which 4 000 t will be classed as radioactive waste. A special aspect of the radioactive inventory and waste will be the tritium, due to the fact that Fugen is a heavy water moderated reactor.

The planning of the dismantling of the reactor and its systems will have to take into account the presence of Tritium both in the heavy water systems and in concrete. The pressure tube design of the core also requires special consideration. During its operational years and just after the permanent shutdown, Fugen has had full system decontamination five times. The systems will be decontaminated before or after dismantling for reducing worker exposure and activity in waste.

Pyrolysis and "depressurised oxygen plasma" methods have been developed for the treatment of ion exchange resin.

For the Decommissioning Engineering Support System (DEXUS), the 3D CAD data of the entire plant has been put into a database and used for visualising the dismantling process. The COSMARD Code, developed by JAERI, is used to evaluate the workload, exposure of workers, waste arising and schedule of the dismantling. A basic decommissioning plan is expected to be ready by the beginning of 2006.

A1.1.12 Tokai 1 NPP, Japan

The Tokai 1 reactor was the first operating commercial nuclear power plant in Japan. It was a 166 MWe Magnox Gas Cooled reactor and operated between 1966 and 1998. It is also the first decommissioning of a commercial nuclear power plant in Japan. After shutdown, the reactor was defuelled under its operating licence and all fuel elements were shipped offsite for reprocessing by June 2001.

The decommissioning project was started in December 2001 and is expected to be carried out over 17 years in three phases. The site would then be in a "green field" status and will be re-used for the siting of a new nuclear power plant. The first phase is 5 years. The first activity was the preparation of the reactor for SAFESTORE by closing all primary system valves to the reactor in December 2001. Conventional facilities will also be removed. During the second five-year phase, the steam raising units and the primary gas ducts outside the reactor building will be dismantled. The reactor itself will be in a SAFESTORE condition during these first two phases i.e. over a period of 10 years. A dose uptake study has shown that the worker dose for decommissioning activities will be at the same level as during plant operation. All reactor structures and associated equipment will be dismantled during the seven-year third phase. This phase will also cover the demolition of the reactor and other buildings after clean-up and a radiological survey. Even after clean-up to a green field status, the land will be continuously controlled as a restricted area of the operating 1 100 MWe BWR Tokai 2.

During the on-going first phase the main activities after the establishment of Safestore have been:

- Sampling of the concrete in the Turbine building. About 75 t of samples were taken for an R&D study on concrete recycling.
- The fuel cartridge cooling pond was cleaned up. The fuel racks were decontaminated, segmented and packed into waste boxes. 2 700 t of pond water was drained through the active effluent treatment plant for discharge.
- The electrical supply was simplified and renewed as was the reactor auxiliary cooling water system. Residual lubricating oil and seal oil was drained and the system washed with hot water and steam.
- The turbine-generator, condenser and associated ductwork have been removed during 2004.

Other equipment planned to be removed during Phase 1 will include:

- Large diameter piping on the reactor building walls such as the main steam lines.
- Equipment in the Fuel Handling Building such as the CO_2 storage tank, gas dryer etc.
- Equipment in the Reactor Service Building such as the diesel generator, feed water pump etc.
- Fuel charge machine, transporter, etc.

A major item of radioactive waste arising from the dismantling of the Tokai 1 reactor will be the 1 600 t of graphite core. In order to prepare the safety case for its burial as low level radioactive waste, 29 graphite samples have been extracted from the core. A special trepanning tool and equipment to use it was designed and used for the purpose. The method was first tested on an inactive mock-up. The samples were about 30 mm in diameter and 50-80 mm long. They have been taken to the hot laboratory for chemical, radioactivity and nuclide tests, mainly to collect the ^{14}C data necessary for making the safety case dose estimation. It is planned to take samples of the steel in the core internals.

The total costs for the Tokai 1 decommissioning project have been estimated at 685 MEuro (2001). Of this amount, 54 MEuro are for radioactive waste management disposal, i.e. over 60%. This underlines the need for reducing the volume of radioactive waste as well as the cost of construction and operation of the final repository.

Many of the technologies necessary for the dismantling/segmenting of the in-vessel details of the Tokai 1 had earlier been developed and used on the JPDR decommissioning project. A number of tests to verify that these technologies can be applied to the Tokai 1 plant are being undertaken by the National Power Engineering Corporation (NUPEC).

A1.1.13 Korea Research Reactors 1 & 2 (KRR-1 and 2)

Korea Research Reactor 1 (KRR-1), the first research reactor in Korea, has been in operation since 1962, and the second one, Korea Research Reactor 2 (KRR-2) from 1972. The operation of both of them was phased out in 1995 due to their lifetime and operation of the new and more powerful research reactor, HANARO (High-flux Advanced Neutron Application Reactor) at the site of the Korea Atomic Energy Research Institute (KAERI) in Daejeon. Both are TRIGA Pool type reactors in which the cores are small self-contained units sitting in tanks filled with cooling water. The KRR-1 is a TRIGA Mark II, which went through the first criticality in May of 1962 and could operate at a level of up to 250 kW. The second one, the KRR-2 is a TRIGA Mark III, which could operate at a level of up to 2 000 kW.

The decommissioning project of these two research reactors was started in January 1997 and will be completed by 2008. The aim of the decommissioning activities is to decommission the KRR-1 & 2 reactors and to decontaminate the residual building structures and the site to release them as unrestricted areas.

KAERI submitted the decommissioning plan including the environmental impact assessment report to the Ministry of Science and Technology (MOST) for the license in December 1998. This was approved in November 2000 after a long review by the Radiation Protection Sub-committee on Nuclear Safety and by the Nuclear Safety Commission – the highest consulting commission related to radiation safety issues in Korea.

A domestic company was selected by open bid as the main contractor to do the decommissioning work. But radiation protection and health physics works was separately contracted to a third company so that the independence was guaranteed. Simultaneously, a detailed work procedure, detail radiation protection guidelines and procedures and waste management procedure will be prepared before starting the practical decommissioning works.

According to the schedule, the practical decommissioning activities will be started in June 2001 by cleaning first the radioisotope production equipment and experimental laboratories in the KRR-2. More seriously contaminated areas such as the lead hot-cell and concrete hot-cell will then follow. The two reactor halls and the reactors themselves will be dismantled from 2003.

All the dismantled materials should be classified by the three following categories: non-contaminated, radioactive material lower than the free release level and material higher than this. The non-contaminated wastes will be disposed of like industrial waste. The second one will be temporarily stored on the site then disposed of after permission from the Minister of the Ministry of Science and Technology. The radioactive wastes will be further volume reduced by decontamination by proper techniques such as washing, cutting, compacting etc., and put into 4 m^3 containers for temporary storage on the site. These will then be transported to the national LILW repository when it is operational, probably in 2008.

A1.1.14 Bohunice A1 Project, Slovakia

The A1 Bohunice Nuclear Power Plant is situated about 2 km from the village of Jaslovske Bohunice. Building started in 1958; the station achieved criticality in October 1972 and started operation in December of the same year. It was a heavy water moderated, CO_2 cooled, pressure tube reactor. Two accidents took place in the facility and in the second (in 1977), fuel overheated and there was leakage of fission products into the primary system and the moderator. A decision was taken in 1979 to shut down the plant and decommission it.

The company, Slovenske Electrarne VYZ was established in January 1996, with the primary goals of reaching a safe store status for A1 and to take responsibility generally for radwaste management and decommissioning. This would cover, apart from NPP A1, also the V1 and V2 VVERs operating at Bohunice as well as the reactors under construction at Muchovce. SE VYZ, which has about 470 employees, has built a radwaste treatment centre BSC, where low and intermediate level waste will be conditioned for acceptance at the shallow land disposal facility at Muchovce. At the BSC, evaporation, cementation, supercompaction and incineration plants are being brought into operation. The non-active tests have been completed. In addition, a bituminisation plant is in commercial operation, for processing the concentrates from the V1 and V2 plants. There is also a vitrification plant for treating the Chrompik solution (see below).

Fuel assemblies damaged in the 1977 accident had been stored for 20 years in Chrompik (Potassium Chromate) solution. Special machines have been designed for piercing the fuel cans, draining the Chrompik, in an atmosphere of Argon (to avoid the risk of explosion). There is also a cutting machine for cutting the fuel into lengths of maximum 5.5 m to accommodate them in the special transport cans.

All spent fuel has been transported off site and sent to the Russian Federation, without any major technical, organisation or legislative problems. The preparation of the last 16 assemblies involved a collective dose uptake of about 400 mSv by 106 persons, due to:

- The high surface dose rate from the fuel cans.
- The high volume activity in the spent fuel (long-term storage) pond.
- The high levels of contamination on the upper parts of the cans.

A 500 l/h evaporator has been installed to concentrate liquid waste into concentrated sludges, which are later cemented. The sludges are liquid enough to be pumped. Waste is conditioned in special containers, which are 1.7×1.7×1.7 m cubes of fibre reinforced concrete. Cementation of the evaporator concentrated is in the cubic containers, into which the sludges, cement and additives are added through metering devices, and mixed. Cementing in cubes is also used for processing ion exchange resins as well as ashes from the incinerator. Only one type of waste is processed in any one cube.

The NUKEM-supplied incinerator has a capacity of 50 kg solid or liquid waste/h, with a maximum heat content of 20 MJ/kg. It is operated batchwise for 20 h, followed by a 4 h afterburn. It is started with propane gas, followed by oil fuel. The dry waste is sorted, packed into 30 l polyethylene or paper bags, about 3.8 kg/bag. The feed is about 10 bags/hour, transported to the incinerator by a vibratory conveyer. The secondary waste is about 4 kg ash/h, which is removed after the 20 h operation (i.e. 80 kg of ash each day). The ash is sent for cementing. There is no heat recovery from the exhaust gases, which pass through two wet scrubbers after temperature reduction, a drier, and HEPA filters, before release to the atmosphere. The secondary waste from the scrubbers is cemented. The release norms from the incinerator chimney are equivalent to the European levels (for CO, dust, etc.). There is no limit at present on dioxins.

The 20 000 kN supercompactor was built in the Czech Republic. It achieves a volume reduction factor of 10 and can treat 10 drums/h.

The waste being treated in the bitumen plant now is liquid waste evaporator concentrates from the V1 and V2 plants. Ion exchange resin is not being conditioned, as there were problems with the drier unit.

The bitumen treatment plant is a thin film evaporator, accepting 80-110 l/h of concentrates and uses 20-30 kg/h of bitumen. The ratio of bitumen to concentrates depends on activity, salt content, pH, etc. and is varied to achieve a final product according to specifications. The plant can accept salt concentrations of up to 200 g/l. Higher concentrations have to be diluted. The concentrates are transported in special (licensed) containers within the Bohunice site, and pneumatically pumped into storage tanks. During the process, they are pumped into operation tank and heated to 80°C before being fed into the thin film evaporator, to which bitumen is fed at 110°C. The product is fed into 200 l drums, with identity labels on each drum. The plant is operated round the clock (24 h) in campaigns, with periodic shutdowns for preventive maintenance.

A vitrification cell has been built for vitrifying, after concentration, Chrompik (Potassium Chromate) solution. The original solution, with 30 g/l salt, is concentrated to about 500 g/l salt. The Chromium is reduced in valency from Cr VI to Cr III. 50 l of the concentrated solution, with an activity of 10^9 Bq/l, is fed into an evaporator and silicon dioxide and other additives are added. The mixture is vitrified in a medium frequency, hot crucible. The molten glass produced is poured into stainless steel canisters. The activity in the glass is about 10^{11} Bq/l.

The cell was taken into test operation in 1997. 12 m^3 of Chrompik solution has been treated, resulting in 229 canisters. To date, the licence covers only solution up to 10^{10} Bq/l. To be treated are solutions up to 10^{11} and 10^{12} Bq/l. At present, Chrompik with activity concentrations of up to 109 Bq/l is being vitrified.

Decontamination to free release levels is performed both with chemical and electrochemical methods. The chemical decontamination is in batches of 200-500 kg in a chemical bath with ultrasonic intensification and is based on formic acid, corrosion intubitor and chelating agent, for use on low alloy steels. About 97% of the metal could be released without radiological restrictions. The required levels for free release and the averaging values were specified in the report from the project.

A1.1.15 Vandellos 1, Spain

The Vandellos 1 plant was a 500 MWe gas graphite reactor of the same type as the EDF St-Laurent-des-Eaux plants. It was shut down after 17 years of operation, after a fire in the turbines in 1989, a level III incident. There was no release of radioactivity, but the fire destroyed the conventional plant and flooded the bottom part of the reactor building.

One characteristic of the Vandellos plant is that the nuclear steam supply system is integrated, with the core and the steam generators contained inside a 19 m diameter, 36 m high (internal dimensions) prestressed concrete pressure vessel. Above the reactor vessel was a refuelling machine that refuelled the reactor on load, thus achieving the very high availability factor of 92%. The reactor building contains, apart from the concrete pressure vessel, also the blowers and other auxiliary equipment. There are other buildings for housing the irradiated fuel, auxiliary electrical power, etc.

The chosen decommissioning alternative is to achieve Stage 2, i.e. dormancy. A dormancy period of 30 years is being planned for. During that period, all radioactivity on site will be concentrated inside the pressure vessel and the vessel will be isolated. At the end of the current project, all buildings on site will have been dismantled, except the reactor vessel and a special protective building round it.

The plant was placed in a safestore situation during April 1991 to October 1994. The staff at the station has been reduced from 315 in 1990 to 110 in 1996. The early activities were connected with the treatment of operational waste, which consisted of solid wastes, irradiated metal (control rods), resins and graphite.

The graphite fuel sleeves, which were stored in 3 silos on site, consisted of about 1 000 t of graphite and 2 t of stainless steel wire, with the graphite containing low activity, long life ^{14}C and the wire containing high activity, relatively short life ^{60}Co. The packaging project packed the graphite sleeves with a vertical tool (manipulator) into baskets, which were loaded into a metallic container (8 mm wall thickness) of suitable dimensions to be placed in concrete at El Cabril, and activated wires were loaded into a high integrity containers.

ENRESA received the authorisation to proceed and took over responsibility for the site from the utility in 1998. The dismantling of the plant was started, beginning with the conventional plant and then continuing with the active part. By the end of 2000, all the conventional and 80% of the active plant had been dismantled.

The reactor pressure vessel, which will be left on site, is of concrete, with 5 m thick walls and top and bottom slabs about 6-7 m thick. The activity content of the vessel is about 100 000 Ci, mostly ^{60}Co. The residual heat equivalent is about 4-5 kW in the graphite and other materials. Part of the Stage 2 concept is the total static isolation of this vessel. The vessel has 1 700 penetrations, the pipes of which were cut, seal-welded and inspected. The covers were insulated with polyurethane foam and various forms of physical protection installed. This total sealing is to avoid condensation in the core area.

The leak-tightness of the vessel was tested by subjecting the vessel to a slight over-pressure of the order of 0.5 kg/cm^2 and to evaluate the leakage over a period of time. The results were very satisfactory, about 18% of the acceptance criteria.

The Vandellos project is very systematically aiming to minimise the quantity of radioactive waste arising. The management of materials emerging from controlled areas is based on a rigorous process of measurement of gamma emitting nuclides and estimation of activity of difficult to measure nuclides. The procedures for this, including the campaign and using conservative judgements, have been submitted to the authorities for approval. An authorisation was issued in the autumn of 2000.

A test plan has been required by the authorities, including not only the measuring devices, but also all the procedures in the process, with an independent quality control by sampling of materials. It is expected that about 90% of the redundant material can be released. Of the 10% remaining, hot spot elimination can reduce the radioactive waste to 2-3% of the candidate material.

More than 10 000 t of material coming from controlled areas have been released as conventional waste taking advantage of the clearance process.

In collaboration with EDF, France, the graphite in the core is being characterised. Fifty-eight samples have been taken in various positions, with a robotic drilling tool. The samples will be measured for impurities, etc. The results will be used for the resolution of the problem of graphite disposal from GCRs.

The Vandellos project has an information centre for the public, which is mobile, and equipped with videos and windows, allowing the visiting public to watch dismantling operations safety, without interfering with the work.

The end of the level 2 project has been reached in the last quarter of the year 2003. The total budget of this level (fuel and waste disposal not included) was about 90 MEUR.

A1.1.16 Taiwan Research Reactor (TRR), Chinese Taipei

The Taiwan Research Reactor at Lung-Tan, Tao-Yuan, (about 200 km south of Taipei) is a 40 MWth, natural uranium, heavy water moderated and light water cooled reactor that operated between 1973 and 1988. It has been decided to partially dismantle the TRR to build, instead, at the same site, a TRR-II, which will be a light water moderated, pool type, multi-purpose research reactor. For achieving this, the partial dismantling will be performed under two phases:

- Phase 1 will cover the removal of the reactor vessel, the various shields in the reactor cavity as well as dismantling of all the redundant systems.

- Phase 2 will deal with the segmenting of the reactor vessel and the management of the waste arising from the decommissioning project.

The reactor had been defuelled and the heavy water removed by 1990. All systems and components within 5 m of the reactor block had been dismantled and removed before 1994. The reactor and systems had been radiologically characterised by sampling and by computer estimations.

The reactor vessel will be separated and lifted from it base and transferred to a specially built dismantling building, where it will be segmented from the top down. The segmenting will be using under water plasma cutting. For this and other cutting purposes, tests have been carried out with several technologies such as:

- Plasma under water.
- Abrasive water jet.
- Electric-discharge-machining (EDM).
- Carbon gouging and mechanical cutting.

Development work has also been performed on cutting platforms and manipulators. This work has also included test on full thickness, full size mock-ups.

As there is a great shortage of storage space for radioactive waste in Taiwan, an interim storage silo is being constructed for receiving the waste arising from the TRR Partial Decommissioning Project. It will have 93 stainless steel lined vaults.

After the removal of the reactor vessel and redundant systems, the TRR building will be cleaned up to release limits (4 Bq/100 cm^2) to allow the construction of the new TRR-II.

The TRR Partial Decommissioning Project has a budget of 850 million NT$ (approximately 25 MUS$).

A1.1.17 Windscale Advanced Gas-cooled Reactor (WAGR), United Kingdom

The Windscale Advanced Gas-cooled Reactor was a 100 Wt reactor that operated between 1962 and 1981. After defuelling its decommissioning to a Stage 3 status was started at the end of 1983. A central feature of the project is the remote dismantling of the active internals of the reactor vessel.

The WAGR project, its time schedule and its execution have been significantly affected by the extensive reorganisation of the United Kingdom Atomic Energy Agency (UKAEA) and the setting up of AEA-Technology, first as a separately operated government owned company and later as a privatised commercial contracting company to sell its scientific and engineering capabilities to industry within the UK and overseas. During the years that this took place, funding to the project, which was on an annual basis, was reduced and some project activities were delayed. The WAGR has been an EU-supported pilot-dismantling project and Nuclear Electric and Scottish Nuclear have made significant financial contributions.

Starting in 1983 as a project to develop decommissioning techniques, it is now demonstrating the cost effective management of dismantling on a large project scale and the safe management of waste arising. Some project milestones have been:

- Dismantling of refuelling machine (by 1989).
- Removal of top bioshield and pressure vessel top dome (by 1992).
- Waste packaging route developed and built.
- Remote dismantling machine (RDM) for vessel internals in place (by 1993).

- All four steam generators removed and transported to the Drigg repository.

During the first years of the project, a waste processing building was erected and a waste route established to it by jacking up two steam generators. All operational waste was processed. The refuelling branches were cut to just above the top dome of the reactor vessel. During 1991, the top dome was dismantled and the refuelling branches were further trimmed to the level of the top of the hot box.

A waste box design has been prepared to meet the requirements of NIREX, the UK waste agency. The waste box is of reinforced concrete preformed and manufactured offsite. After being filled with waste, grout is injected into the interspaces. After curing, reinforcement is arranged over the top and a concrete lid is cast on, thus producing a monolithic final product. A prototype waste box has been used to test the operability of the waste route.

The waste box will be used both for LLW and ILW. Tests showed that ^{137}Cs leaked from such boxes. Activity was leached out during grouting and transported out through cracks in the box. It has been decided to use polymer-modified cement to solve this problem. This has been approved by the authorities.

Single piece disposal of each of the four WAGR heat exchangers was chosen after earlier consideration of sea disposal, decontamination, dismantling etc. The chosen method involved:

- Preparation of each heat-exchanger for liftout.
- Preparation of the containment building to allow liftout.
- Liftout of each heat-exchanger.
- Transport to the Drigg waste disposal site (16 km).
- Clean up and reinstatement after liftout.

The transport was successfully carried out using a 96-wheeled modular trailer. Responsibility for the heat exchangers was transferred from UKAEA to BNFL (owners of Drigg) when they were lifted off the trailer.

The heat exchangers were placed in concrete vaults at Drigg, each of them grouted both internally and externally, to satisfy the Drigg requirement of monolithic waste. The heat exchanger transport project was a very visible one and very successful from the public relations point of view.

Originally it had been planned to use the RDM with an oxy-propane thermal torch with iron powder injection to segment the hotbox. It was later seen that the use of this high-temperature "aggressive" method could spread radioactivity (especially ^{137}Cs) through the rest of the reactor and complicate their later dismantling. The new strategy is to dismantle this relatively low dose rate component by a series of semi-remote and manual operations.

The alternative strategy suggests:

- The use of plasma cutters instead of oxy-propane with powder injection at a number of locations.
- Removal of the fuel element guide tubes at an early stage, thus removing the largest suspected source of ^{137}Cs.
- Other parts to be cut by a grinder (held by the RDM) and by a hydraulic shear.

The future scheduled work on the WAGR is:

- Loop tube removal (by 2000).
- Neutron shield dismantling (by 2001).
- Graphite core and restraint structure removal (by 2003).

A1.1.18 Prototype Fast Reactor (PFR), United Kingdom

The Prototype Fast Reactor is one of several nuclear facilities at the UKAEA Dounreay site, in the north of Scotland. At the site, are also:

- A shutdown pool type test reactor.
- A reprocessing plant.
- The Dounreay cementation plant and other waste management facilities.
- A low-level waste disposal site.
- The Dounreay shaft, with intermediate level waste.

The PFR was a 250 MWe (600 MWt) sodium cooled fast reactor that was in operation between 1975 and 1994, when it was shutdown for decommissioning. The closure was announced 6 years before the final shutdown. A decommissioning manager and team were in place two years before shutdown. The aim of the current decommissioning phase is to achieve a safe storage Stage 1 status.

The entire primary circuit of the reactor is contained in a 12.3 m diameter 1.5 m deep stainless steel vessel. There are three pumps for circulating the 900 t of primary sodium coolant through intermediate heat exchangers. These intermediate heat exchangers are part of a secondary coolant system also containing sodium. The secondary sodium has been drained and allowed to solidify in a tank farm, while the primary sodium is maintained in a molten condition by operating the primary pumps.

The first major task after shutdown was the removal of all fissile and fertile material from the vessel. As each fuel element removed from the reactor had to be replaced by a dummy element (without fuel), a whole dummy core had to be manufactured before the shutdown of the reactor. The defuelling of the reactor took about two years.

One of the main activities in the decommissioning project is the disposal of both the primary and secondary sodium. For this, a sodium disposal plant (SDP) has been built by a consortium with NNC as the prime contractor with Framatome and AEA Technology as partners. The inactive commissioning programme is nearing completion. It has been confirmed that the plant is according to design: individual systems have been tested and powered individual system testing is almost complete.

In the next phase of commissioning, 45 t of clean sodium will be processed. The process is basically the conversion of Na to NaOH by controlled exposure to water and then the neutralisation of the NaOH by the addition of hydrochloric acid giving sodium chloride solution, which will be discharged to the sea after clean up of cesium. The next phase will optimise the NaOH and the neutralisation processes and produce a report to the Safety Working Party.

The contract for sodium disposal is in two parts: the liquid metal supply (LMS) to the SDP and the destruction of the liquid metal (SDP). The LMS involves the extraction of sodium from the reactor vessel, the irradiated fuel cave, the tank farm and the drain lines as well as drainage of sodium and NAK from the intermediate heat exchangers.

Other items of interest:

- The primary sodium is heated by electric heaters to keep it molten. They will no longer be needed when pumping starts.
- The 1 000 t of primary sodium contains only about 2 g of ^{137}Cs. It is uncertain how many ion exchange columns are necessary to clean up the sodium chloride solution.
- The normal material for tanks at the site is stainless steel, unsuitable for the hold up tank for sodium chloride before discharge. So a special tank has had to be procured.

A1.1.19 JEN-1, PIMIC Project, Spain

Under the responsibility of the owner of the research centre (CIEMAT), Enresa, over the next several years, will carry out the dismantling of several facilities and installations, including an experimental nuclear reactor that was phased out a few years ago. The Regulatory body has now approved the Decommissioning Plan (2005) and the Environmental Ministry has approved the Environmental Impact Assessment (2005). The municipal licence was issued by the end of February 2006. The global authorization is expected before the end of November 2006.

The general objective is D&D of redundant facilities, upgrade of other buildings and facilities, restoration or affected areas and land, and to generally improve the safety culture.

The D&D activities are broken down into subprojects:

- Reactor building and associated systems.
- Reprocessing plant.
- Liquid radioactive waste conditioning plant.
- Liquid radioactive waste storage.

The project will include the establishment of several auxiliary installations including:

- A waste clearance centre.
- Waste characterization and storage centres.
- Waste conditioning centre.

The project is scheduled to be complete by the end of 2009 and cost a total of >22 M €.

Some of the unique attributes to this project include:

- Location – downtown Madrid on University Campus.
- Space limitations – confined to fenced boundary.
- Interactions with operating Research Centre.
- Social impact and visibility are very important.
- Great variety of stakeholders.

A1.2 Completed reactor projects

A1.2.1 Gentilly-1, Canada

Gentilly-1 was a heavy water moderated, direct cycle, boiling light water cooled prototype reactor that was shut down in 1979 after 15 years of operation. It was placed in a storage with surveillance state (Stage 1) in 1986, and it became the Gentilly-1 Waste Storage Facility thereafter.

During the project to place the plant into the "storage with surveillance" state:

- A dry fuel storage facility was constructed for the spent fuel.
- The service building had been cleared of all equipment and decontaminated, as were parts of the turbine building.
- Waste was stored in parts of the turbine building and in the reactor building.

The service building, which had been cleared and decontaminated, has been converted by the new owners, Hydro-Québec, into an office building including a training centre, complete with a full size simulator for the adjacent Gentilly-2 NPP. The former spent fuel pool is now used as a calibration facility.

Low-level waste generated during the decommissioning project had been stored in parts of the turbine building. This has now been relocated in the reactor building. Asbestos insulation removed from the building during the initial decommissioning activities was shipped off site. The roofing of the turbine building has been repaired.

The cost of maintaining the Gentilly-1 site before the decommissioning project has been C$ 10 million per year since 1979, when it had been shut down. Placing the facility in its current state had costed a total of C$ 25 million over two years.

Since 1986, the facility has been in the "storage with surveillance" state during which the facilities, remaining operating systems and any remaining radioactive areas have been routinely monitored and maintained. The annual costs for the inspections and surveillance are of the order of C$ 500 000 (2000).

A1.2.2 Nuclear Power Demonstrator (NPD), Canada

The NPD was the 25-MWe prototype for the CANDU-type reactor which operated from 1962 to 1987. The decommissioning alternative chosen was the same as for Gentilly-1, i.e. "storage with surveillance state".

During facility shutdown, all nuclear systems were drained and sealed. All fuel was shipped off site for dry storage in concrete canisters at Chalk River Laboratories. Operational wastes were shipped to Chalk River for disposal. At the beginning of the "storage with surveillance" state, there were approximately 2×10^{15} Bq on site, mostly in the reactor vessel.

The unrestricted access areas were thoroughly decontaminated to radiation levels of under 2.5 μSv/h. During the dormancy period, the turbine, generator and auxiliaries were dismantled and sold. During the early 1990s, ancillary facilities were removed from the site.

The NPD decommissioning project to implement the "storage with surveillance" state was budgeted at C$ 18.5 million. Since 1988, the facility has been in the "storage with surveillance" state during which the facilities, remaining operating systems and any remaining radioactive areas have been routinely monitored and maintained. Site security is monitored from Chalk River Laboratories and complemented with periodic on-site inspections. The annual storage with surveillance costs are of the order of C$ 0.3 million compared to C$ 14 million before decommissioning.

A1.2.3 Rapsodie, France

The sodium-cooled fast-breeder reactor Rapsodie operated at 20 MWt and later at 30 MWt. It achieved criticality in 1967 and was finally shut down in 1984. The project to put the facility in a Stage 2 decommissioning status was started in 1987.

The reactor vessel was emptied. For this, the fuel and blanket assemblies were removed from the core, washed and sent to Marcoule for reprocessing. The steel/nickel dummy elements were lifted out and are in interim storage on site. The sodium in the primary systems was drained. The system was washed with ethylglycol and decontaminated with nitro-sulphuric acid process with Cerium IV. The systems were then dismantled and about 70 t of stainless steel from the decontaminated primary loop were sent to the INFANTE facility in Marcoule for melting and use for making iron containers.

The reactor block was then sealed. The reactor vessel was complemented by an upper closure head, constituting a first leak-tight barrier. The outer concrete enclosure was completed with steel caissons on the six sides of the reactor plant, thus forming a second barrier.

The main activity for terminating the Stage 2 decommissioning of Rapsodie was the destruction of 37 t of the sodium coolant. The destruction process, developed by CEA, was a controlled sodium water reaction producing concentrated sodium hydroxide (soda). The 37 t of sodium were destructed in the purpose-built DESORA rig during 16 weeks, resulting in 150 m^3 of 10 M of soda, which will be transported to a COGEMA plant at La Hague for liquid effluent treatment.

On 31 March 1994, a residue of 600 l of sodium was being treated with heavy alcohols for producing a stable salt, when an explosion took place causing the death of one engineer and injuring four others. The explosion occurred in a tank in the gallery outside the containment. The heavy alcohol process had been used earlier for washing the primary system.

The CEA established an internal inquiry commission. A judicial inquiry is now in progress.

A1.2.4 G2/G3 Reactors, France

G2 and G3 were two 250 MWe gas-graphite reactors that operated between 1958 and 1980. In each reactor, the core with reflector and shield plates is located within a prestressed concrete pressure vessel, while the four steam generators and associated primary cooling circuits are outside the pressure vessel. This arrangement has made the plants suitable for Stage-2 decommissioning, where the external cooling circuits and steam generators will be dismantled, while the core and other internals will be enclosed in the concrete pressure vessel.

The external cooling circuits consist of about 1 500-2 000 t of carbon steel in each reactor. The direct disposal cost for this steel was estimated to be about FF 240 million. After studies, it was decided to wash the interior of the systems with high-pressure water and then melt the piping for recycling the metal.

A decision was taken in October 1990 to build a melting facility (INFANTE) at the G2/G3 site. An electric arc furnace was chosen because it was considered safer from the effects of possible water inclusion in the piping and also because it would allow a larger lid opening than an induction furnace. It has a 15-t/charge capacity. Pipes up to a diameter of 1.6 m can be loaded directly, saving considerable cutting costs. Both carbon and stainless steel have been melted.

The melting results in 25 kg ingots or 4 t blocks, which are monitored for radioactivity. The ingots and blocks are stored in the facility awaiting an agreed very-low-level waste disposal repository or a recycling project within the nuclear industry.

After inactive and active tests during late 1991 and early 1992, operations started in April 1992. By the middle of 1994, the contaminated steel scrap from G2/G3 had been melted at INFANTE and the melting facility was used for treating steel scrap from some other CEA facilities.

The CEA has studied various possible ways of using the material resulting from the melting of the contaminated scrap at activity levels higher than releasable. One was the production of waste containers using the “integral workform” principle. Here the cast iron was poured into annular moulds of sheet steel (cylindrical) where the cast iron solidified between the outer and inner steel sheets forming an integral shielded container. Pre-machined inserts had been welded into the sheet steel mould, thus avoiding the need for post-casting machining for lifting points etc. The outer surfaces of the mould were not contaminated and therefore the containers could be handled comfortably. 150 such containers were produced.

The *Société des techniques en milieu ionisant* (STMI), a subsidiary of EDF and CEA, was appointed Architect/Engineer for the G2/G3 site with CEA’s UDIN Department as the operator. The reactors G2 and G3 have formally been placed in a Stage 2 dormancy status. The dormancy period is expected to be between two and three decades.

After the Stage 2 dormancy status has been reached, the INFANTE plant will be decommissioned and dismantled. A new modern induction furnace melting plant has been built by SOCODEI.

A1.2.5 Kernkraftwerk Niederaichbach (KKN), Germany

The Niederaichbach nuclear power plant was a 100 MWe prototype, heavy water moderated, and carbon dioxide cooled reactor. It was shut down in 1974 after having produced the equivalent of 18 full-power days, due to steam generator problems.

A safe enclosure licence was granted in 1982. The licence for complete decommissioning to Stage 3 was granted in 1987, after many years of litigation, public hearings and appeals. Decommissioning on site started in 1988 with the removal of inactive and later contaminated components.

The dismantling of the highly active components of the core region of this vertically oriented pressure tube reactor was carried out with a high precision remotely operated rotary mast type manipulator with suitable tools attached.

The main sections of this core region were:

- The upper neutron shield.
- The 351 pressure tubes.
- The lower neutron shield.
- The moderator tank.
- The thermal shield.

Many techniques were used in the cutting and dismantling of the core components. Among those utilised were:

- Grinding.
- Plasma torch.
- Disc cutters.
- Screw removal.
- Band saw.

The remote dismantling of the core region and segmenting these components took place between November 1990 and March 1993.

These components amounted to 522 t with a total activity of 8.6×10^{12} Bq. They were packaged into 139 containers ready for disposal at the Konrad repository when it becomes operational. About 20 per cent of this metal was below 200 Bq/g. This fraction was sent to the Siempelkamp melting facility for recycling within the nuclear industry.

The next project activity was the removal of all activated concrete structures during the period April-November 1993. These structures were, in addition to the biological shield, the upper support ring, the walls of the coolant distribution room and the wedge areas as well as the lower support ring. Hydraulic and pneumatic jackhammers were used for activated concrete removal, in addition to an electrical excavator with a rock chisel. In certain areas, the concrete was cracked by controlled blasting, then removed and packed manually.

All the surfaces in the building were then decontaminated, after which the release measurements were started. The release levels were:

- 0.37 Bq/cm^2 for β- and γ-emitters;
- 0.37 Bq/g for β- and γ-emitters;
- α-emitters were not encountered during the intensive characterisation work that has been done determining the key nuclides of each decommissioning phase.

About 200 000 measurements were made by project staff. These were checked by about 10% verification measurements by the inspection authority and some more (2-3%) by the environmental authorities. The site was released from the Atomic Law in August 1994.

Following the release of the site, conventional demolition could be started in October 1994. The 130 m high stack was demolished in January 1995. The site attained “green-field” conditions during autumn 1995.

The decommissioning machine, which was used for the remote dismantling of the radioactive details of the core region, was decontaminated by sand blasting. It could not therefore be reused and was scrapped.

A1.2.6 Heissdampfreaktor HDR, Germany

The Heissdampfreaktor (HDR) was a 100 MWt, nuclear superheated reactor plant that operated only for the equivalent of 5 full power days. It was shut down in 1971 and the plant was utilised for various safety related experiments between 1974 and 1992.

The aim of the project was to completely dismantle the facility to establish "green field conditions" and was executed under 3 subsequent sub-licences. Due to the short operating life of the reactor, the activity inventory was small (about 2.2×10^{10} Bq) and the ambient dose rates were low.

Under the first sub-licence (which was an extension of the operational licence), the experimental equipment was dismantled. A total of about 550 t of metals was removed, most of which could be recycled without radiological restrictions.

The reactor systems, including the reactor pressure vessel, as well as other plant components were dismantled under the second sub-licence. The only items remaining were certain infrastructure systems, such as ventilation.

The third sub-licence covered the decontamination and removal of the concrete structures inside the reactor containment, including the biological shield, as well as release measurements on these and the other buildings in the plant. The TAG visited the HDR site near the end of the work under this third sub-licence.

One characteristic feature of the HDR containment was the annular gap between the inside of the containment and the inner concrete structures. Condensation during blow-down tests might have caused the concrete surfaces to be partly contaminated. "Earthquake" simulations had caused surface cracks, allowing the penetration of contamination. All such surfaces had to be decontaminated to 0.475 Bq/cm^2 for ^{137}Cs.

Controlled explosion techniques were used to dismantle the activated concrete structures as well as certain floors in the containment. The other concrete structures were decontaminated and dismantled from the top of the building, level-wise. The inner surfaces were subjected to clearance measurements. The wall structure was cut in about 30 segments, each one being felled into a horizontal position for decontamination, if necessary, and clearance measurements.

The HDR decommissioning project was completed in the middle of 1998, more than a year earlier than planned at a cost of DM 99.7 million.

A1.2.7 Japan Power Demonstration Reactor (JPDR), Japan

The Japan Power Demonstration Reactor was a 90 MWt boiling-water reactor that was in operation from 1963 to 1976. The decision was taken to decommission it to a Stage 3 status in order to:

- Gain experience of dismantling.
- Develop/demonstrate decommissioning techniques.
- Assemble data on various aspects of decommissioning.

The decommissioning project was conducted in two phases:

- A five-year Phase 1 starting in 1981, during which an extensive research and development programme was conducted on the technologies required for decommissioning.
- A Phase 2, carried out during 1986-1996, during which these technologies were implemented to dismantle the JPDR to Stage 3 green-field conditions.

One of the main aims of the R&D programme of Phase 1 was to develop remote cutting methods to minimise the radiation exposure to workers. Radioactive components and structures were removed in the early stage of the dismantling activities, and the remote dismantling techniques developed in Phase-1 programme were put to practical use in the dismantling activities.

The reactor internals were removed by the underwater plasma arc cutting system. The plasma torch was operated in most cases by a mast type manipulator. Otherwise, the master-slave robotic manipulator was used for the plasma torch to demonstrate and verify its newly developed robot technology. First, each reactor internal was removed from the reactor pressure vessel (RPV) wall; the cut piece was then transferred under water to the spent fuel storage pool through the canal. These pieces were cut into smaller segments suitable for packaging using another under water plasma arc cutting system.

After removing the reactor internals, the piping connected to the RPV was dismantled using the rotary disk knife, shaped explosives and conventional cutting tools. Then the RPV was dismantled using the under water arc saw cutting system. Before assembling the under water arc saw cutting system, a cylindrical water tank was temporarily installed in the space between the RPV and the biological shield. The tank was filled with water for cutting the RPV under water.

For removing the biological shield, the diamond sawing and coring system was applied to dismantle upper part of the activated inward protrusion of the JPDR biological shield. The lower part of the inward protrusion was dismantled using the abrasive water jet cutting system. After removing the inward protrusion, radiation levels in the reactor cavity were so low that workers could approach the cavity. The rest of the biological shield was dismantled by using controlled blasting. Vertical charge blasting was used for demolishing the inner portion and horizontal charge blasting for the outer portion. The wastes from the outer portion were disposed by near surface burial at JAERI's site as a demonstration test. The other wastes were put into containers which were stored in the waste storage facility.

In parallel with the dismantling activities in the reactor building, components in auxiliary buildings such as turbine building and radwaste building were dismantled using conventional techniques, such as band saw, reciprocating saw, oxyacetylene torch, and plasma torch. Large components such as the pool lining and the turbine were cut into small segments and stored in the containers.

Information about the JPDR dismantling activities was collected and accumulated in the decommissioning database. This database was used for:

- Managing ongoing JPDR dismantling activities.
- Verifying the Code Systems for Management of Reactor decommissioning (COSMARD).
- Planning future decommissioning of commercial nuclear power plants.

As an example of the analysis for utilising the database for future commercial plant decommissioning:

- The ratio of manpower expenditure to the weight of dismantled components was evaluated to be 500-2 000 man-hours/t in remote dismantling procedure for highly radioactive components, compared to 10-100 man-hours/t with manual dismantling procedure in the reactor building. The remote dismantling systems were proved to be effective for general components to minimise the radiation exposure of workers, which was kept to a collective dose of approximately 300 man-mSv.

- After the removal of the components and systems, the inner surfaces of the JPDR buildings were decontaminated using a number of techniques, including scabblers, needle guns and concrete planers. The total area to be decontaminated and surveyed (radiologically) before release is 12 000 m^2. The buildings have been approved for release by the authorities and later demolished by conventional techniques. The project to decommission JPDR to Stage 3 green fields was completed by the end of March 1996.

Based on the experiences from the JPDR project, a new R&D programme was initiated at JAERI, including:

- Decontamination techniques.
- Radiation measurement.
- Remote dismantling techniques.
- Systems engineering for decommissioning.

In development of decontamination techniques, flow abrasive and laser induced chemical decontaminations were selected to study on their capability. The laser induced chemical decontamination tests indicated the possibility to reduce spot contamination from 400 Bq to non-detectable level using gel-type chemical reagent. To achieve high sensitivity in radiation measurement under natural background conditions, the method to discriminate β-rays from counting of both γ- and β-rays was applied using a double layers gas flow type counter. It was confirmed that the minimum detectable level achieved was approximately 0.1 Bq/cm^2 for ^{60}Co contamination in 60 seconds counting time. Two kinds of detectors were fabricated and these were attached to the movable machines for measurement of radioactivity on building surfaces and piping embedded in building structures. As for remote dismantling techniques, dual arm manipulators were manufactured to study on automated remote dismantling work based on computer simulations. The dual arm manipulators are controlled by the packages of robotic language prepared by computer simulations. The applicability of automated control system was examined by dismantling mock-ups of components. In systems engineering, project management tools using expert systems and database on dismantling activities have been developed. The systems are intended to be applied to estimation of radioactive inventory, project resources, worker dose, and scheduling in a decommissioning project by referring data obtained in past experience.

The R&D programme was completed by March 2001. It is expect that the developed technologies and data will be applicable to further decommissioning of nuclear facilities in Japan.

A1.3 Fuel facility projects in progress

A1.3.1 Eurochemic Plant, Belgium

The Eurochemic reprocessing plant at Dessel was originally owned by a consortium of 13 OECD countries and operated between 1966 and 1974. The plant was decontaminated after shutdown in order to reduce the standby costs. The plant was later transferred to Belgian ownership. Belgoprocess was created in 1984 to take over responsibility for the site. The decision to decommission to a Stage 3 status was taken in 1986.

A pilot project was carried out, to test techniques and costs as well as to train personnel on two storage buildings. Since 1989, the main project of decommissioning the reprocessing plant has been proceeding. The main process building (80 m long, 27 m wide and 30 m high) is currently being decommissioned.

The decommissioning activities have concentrated on:

- Cost minimisation by actions to reduce standby costs and by decontamination to unconditional release levels.
- Using commercially available technology and adapting it for use in a nuclear environment.
- Achieving acceptable conditions for working in an alpha contaminated environment.

Some of the specific technical achievements of the Eurochemic decommissioning project have been:

- Developing a decontamination system for concrete surfaces with in-depth contamination. A machine was developed with 4 scabbler heads, which could be used on floors, walls, and ceilings.
- Later, the development of an alternative “shaving” process, using a diamond tipped rotary head, which gave a smoother, more easily measured surface and which reduced the secondary waste (compared to scabbling) by 30%. Shaving was first applied on floors and later on walls. Later a hand-held version has been developed.
- An industrial scale dry abrasive blasting machine for the decontamination of contaminated metallic profiles and plates. Operational activities started in 1996 and at the end of September 2004 about 904 Mg of contaminated material has been treated. 219 Mg of this material has been released having been measured twice by the in-house health physics department. About 615 Mg of metal, representing surfaces that cannot be measured due to their shape, have been packed in drums or bins and were melted for release in a controlled melting facility.
- In the same installation, also about 237 Mg of concrete and heavy concrete blocks were decontaminated. 210 Mg (89%) of this material was unconditionally released having been monitored twice by the in-house health physics department or after other treatment in the crushing and sampling installation. The unit cost for abrasive decontamination proved to be about 45% of the global cost for radioactive waste treatment, conditioning and disposal of the same material.
- A ventilated suit for working in (especially α-) contaminated areas. The suit is provided with both cooling and breathing air,
- Monitoring and control of hand/arm vibrations associated with manual work.

A recent development has been in the field of clearance of concrete. In view of the final demolition of the main process building, a specific clearance methodology was evaluated. Application of surface monitoring and core sampling as was used on the earlier pilot project is difficult for the main process building. This is mainly due to the penetration of contamination to greater depths, to the very large surface areas that have to be monitored, the large number of core samples that have to be taken and the difficulty to prove that these core samples are representative for the remaining building structures.

An alternative methodology was therefore proposed considering one complete measurement of all concrete surfaces and a controlled demolition of the building structures, after removal of the pipe penetrations. The remaining concrete (maximum input dimension $40 \times 40 \times 20$ cm^3) is then crushed to pieces with maximum dimensions of 40 mm, and industrially sampled for monitoring in accordance to relating international norms. The licensing documents for an industrial scale plant were prepared and approved. Orders were placed and the entire crushing installation, the metal separator and the transport and filter systems were delivered in September 2000. Operations of the facility were started in June 2001. At the end of September 2004, 1 934 Mg of concrete were monitored. All this material will be unconditionally released and removed from site after analyses and agreement with the in-house health physics department and the relevant authorities. The material is further used in conventional road construction.

A1.3.2 Building 204 Bays Decommissioning Project, Canada

The 204A and 204B Bays are storage pools in Building 204 at Chalk River Laboratories associated with the NRX Reactor. They are at or above ground level and were put in operation in 1947 for storing or transferring fuel and irradiated components. Originally they were a continuous set of bays and trenches extending to the Building 220 fuel processing facility. After alterations in 1958-59, they were separated by a sand-filled section of trench and concrete dams into two areas A and B. The 204A Bays remained operational until the NRX Reactor shutdown in 1993. The 204B Bays have remained water filled but unused since 1959.

When the project joined the Co-operative Programme, the 204A Bays contained water, approximately 8 m^3 of sludge and algae, and operational components. The sludge and algae have since been removed through a vacuuming process. The 204B Bays contain water and a build-up of sludge and algae but no operational components.

204A Bays

Water	800 m^3	4×12^{12} Bq	98% Tritium, 1.3 g U, etc.
Components, etc.	20 t	6×10^8 Bq	44% fission products 42% activation products 14% actinides

204B Bays

Water	400 m^3	3.4×10^{11} Bq	75% ^{137}Cs, 20% ^{90}Sr, 5% other
Algae/sludge, etc.	9 m^3	9.0×10^{10}	88% fission products, 10% actinides
Components, etc.	None		

The aim of the Bldg 204 Bays project is to clean out the Bays, at present containing algae/sludge, and equipment; and then to store them in a dry state, with only fixed contamination and requiring minimum maintenance. At present, the 204 A Bay leaks at the rate of 4 m^3 of water per day. Both the A and B Bay walls are in equilibrium. A sudden stoppage of the leak could cause a catastrophic failure of the walls.

Detailed decommissioning plans were submitted to the regulators for both A and B Bays in 1999 and a revised version re-submitted in 2001. These plans covered the removal of integral components and debris as well as water treatment. The project will be executed in a phased approach aimed at emptying the 204A Bays first (eliminate the leak) and then proceeding with the 204B Bays.

The project laid down significant efforts in negotiating resolution of issues raised by the regulator on the licensing documentation, particularly on the scope of the environmental assessment. These issues have now been resolved and the documentation is revised and re-submitted.

In the meanwhile, there has been little progress in technical areas. Major project work is awaiting licensing approvals. There is a continuing process of cleaning the bays of sludge and debris (algae growth is a continual problem) and radiological characterisation in support of the planned activities. Special tools and apparatus for waste handling and removal (overhead cranes, storage containers, and flasks) have been designed and manufactured or purchased.

A1.3.3 AT-1, France

AT-1 was the pilot plant for reprocessing fast breeder fuel from Rapsodie and Phénix. It was shut down finally in 1979. The decommissioning unit of the CEA, UDIN, took over the plant in 1982. The first three years were taken up by planning and studies. Dismantling started in 1984.

The decommissioning has taken place in five steps:

- Alpha cells.
- Small beta-gamma cells.
- High-active cells using the ATENA.
- Storage and fission product cells.
- Cleaning out of the plant.

Dismantling started with alpha cells and glove boxes in 1984. This continued during 1985 and 1986 and was taken up again in 1990. During 1987-89, the peripheral equipment was dismantled, the remote dismantling machine, ATENA, was procured and installed. Using the ATENA, which has a 6 m long articulated arm, the highly active cells were dismantled during 1990-92. This has been one of the major achievements of the project.

The ATENA carrier originally started operations with manipulator MA23, which was later replaced by the heavier duty RD500. However, the RD500 suffered a cabling failure in the workshop. So ATENA reverted to the MA23, while the RD500 was being repaired. ATENA had a high reliability, while the manipulators were not so good in this respect. The ATENA dismantling machine, being very site-specific, cannot be re-used and will be sent to Aube repository, for final disposal.

During the period of the execution of the project, there have been great changes in the French approach to releasing nuclear sites as well as materials from such sites. Officially, at present, there is no release level (activity concentration level) for material from areas (zones) classified as nuclear. So the approach has been to clean up to very low activity levels, re-zone the building by declaring it as a conventional zone and then treat it as conventional waste.

In the case of AT1, the maximum residual contamination limits for cleaning up were set to 1 Bq/g for alpha and 100 Bq/g for beta-gamma contamination, measurement is be made by sampling.

Following this was the work on cells with limited access such as those for fission product storage and finally the removal of ATENA, dismantling its maintenance cell and decontaminating a number of cells to release levels. Decontamination was by sand and shot blasting, using a robot carrier arm or manual application depending on location. The remote carrier arm could also be used for holding a post-decontamination measuring head. Some volumetric contamination has been found in the concrete walls of the hottest cells, requiring several centimetres deep scabbling before the final dismantling operations.

An example of such contaminated areas was in Cell 905, where there had been leaks in the dissolver leading to deep contamination of the concrete floor. A remote controlled BROKK machine fitted with a rotary cutter in a drill was used to remove the floor to about 10 cm depth, extending at places to 20-30 cm in depth. The ceiling which was made up of removable steel slabs, had to be stripped using grinding machines. 47 samples of thick wall concrete were taken after clean up. These showed an average activities of 5.65×10^{-2} Bq/g (alpha) and 4.87×10^{-2} Bq/g (beta-gamma), i.e. far below the suggested maximum residual contamination limits, thus qualifying the cell for rezoning as a non-nuclear zone.

Clean up operations on the AT1 was completed by the summer of 2001. The residual contamination on site is less than 37×10^{6} Bq (1mCi), which is the threshold for Installations Classified for Environmental Protection.

A1.3.4 Radiochemistry Laboratory, Basic Nuclear Facility 57, France

Basic Nuclear Facility 57 is located at the French Atomic Energy Commission Nuclear Research Centre at Fontenay-aux-Roses, near Paris. This facility was used between 1961 and 1995 for research and development work on reprocessing spent fuel, production of transuranic elements, tests and analysis relating to these activities.

Basic Nuclear Facility 57 consists of three buildings, Building 18 (plutonium chemical laboratory) and Buildings 91 and 54. Building 18 has an area of 8 000 m^2 (104×78 m). This building is divided into four fire zones making up units. Each unit has an area of 2 000 m^2 (40×50 m) and is composed of a hall ($\leq$ 675 m^2) four laboratories (141 m^2 each) and several annexes. The shielded lines (α, β and γ) are in the halls. The laboratories are used for bench studies in hoods and glove boxes.

Buildings 91 and 54 were mainly used as test bays for chemical engineering pilot installations, using non-radioactive or low-level materials (natural and depleted uranium), representative of the processes developed as part of the activities in Building 18, and as an interim storage area for equipment and materials coming from Building 18. They have a surface area of 1 600 m^2.

The R&D experiments carried out in Basic Nuclear Facility 57 were finished in June 1995. Decommissioning of Basic Nuclear Facility 57 will require two separate operation phases:

- The clean-up operations, already underway.
- The dismantling operations, which could start after a formal administrative authorisation (licensing decree).

The aim is to reach IAEA decommissioning stage 3 without demolishing of civil works. Subsequently, Basic Nuclear Facility 57 will be struck off the list of Basic Nuclear Installations. In the second phase, it will be demolished at the same time as the other nuclear installations in the centre (RM2, SAR, STEL, etc.).

The clean-up operations started in 1995 and are planned to be completed by 2002. It consists of the following tasks:

- Removal of nuclear materials.

- Removal of radioactive sources.
- Treatment and removal of aqueous effluents.
- Treatment and removal of organic effluents.
- Treatment and removal of waste.
- Pumping out plutonium and transuranic contaminated solvent.
- Flushing and decontamination of tanks and pipes.
- Section cleaning of Building 18.
- Section cleaning of Building 91 and 54.

One of the priorities given is waste minimisation (LLW and HLW).

The clean-up operations, including studies and licensing, started in 1995 and ended in 2005. The main dismantling operations starting in 2005 include:

- Shielded line containment removal.
- Glove box cutting-up and removal.
- Liquid effluent storage tank cutting-up and removal.
- Effluent system disconnection.
- Civil works clean-up.
- Exhaust system dismantling (air supply, laboratories, glove boxes, lines, etc.).

The clean-up operations were 81% complete on 31 December 2004. Progress with the individual tasks was as follows:

- Removal of nuclear materials: 100%
- Removal of radioactive sources: 76%
- Treatment and removal of aqueous effluents: 59%
- Treatment and removal of organic effluents: 80%
- Treatment and removal of waste: 84%
- Pumping out plutonium and transuranic contaminated solvent: 47%
- Flushing and decontamination of pond and pipes: 53%
- Cleaning of Buildings 18, 91 and 54: 84%

Aspects of technical interest in the project are:

- A Ce IV based decontamination method with gadolinium buffering to avoid Pu criticality.
- A 100 kg capacity hydraulic arm for use with a remote dismantling machine.
- The use of ISOCS gamma spectrometry.

The decommissioning of the radiochemistry laboratory and buildings 91 and 54 is estimated to cost MEuro 257 (2000).

A1.3.5 ATUE, France

The facility was used for the recovery of enriched Uranium with the facility in operation between 1965 and 1996. During the 30 years of operation more than 500 t of Uranium were recovered. The facility was also used for process support to industry and in 1975 the dry process for UF_6 was designed.

A special licence for clean up was obtained in August 2000. The decommissioning project objective is to achieve Stage 3 except for any civil works demolition. The will be free of any radiological constraints.

Decommissioning is to be undertaken in two phases:

- Final shutdown and post operational clean out (POCO): all reactants and uranium will be removed, dismantling of the process equipment.
- Process decommissioning and concrete clean up: all process deconstruction, general concrete clean up to allow non-radioactive waste mapping leading to the delicensing of the building.

The decommissioning project with the site delicenced will be completed in 2007. Building demolition will commence later in 2007.

The total cost of the project is estimated at 30 MEuro.

A1.3.6 ELAN IIB, France

ELAN IIB was a plant at the COGEMA site in La Hague used to manufacture ^{137}Cs and ^{90}Sr sealed sources in high activity shielded cells. The plant operated between 1970 and 1973 when it was shutdown for economic reasons. Early decommissioning work to place the facility in a Safestore status with surveillance was carried out between 1981 and 1996. In 1996, the current decommissioning project was started with a study being undertaken. The available documentation was collected. The ventilation system was refurbished and a new fire protection system installed. CEA/DEN/DPA is the facility owner. The prime contractor is CEA/DEN/DDCO/SPRO (formerly the UDIN department). The nuclear operator of the site is Cogéma. There is an agreement between Cogéma and CEA regarding the ELAN IIB plant.

All the cells of ELAN IIB except for cell 900 have been emptied. Cell 900 still contains process systems consisting of tanks, pipes etc. It is not known whether the tanks are empty or full. Nor are the chemical and radiological properties of the contents are known. So the first activity of the project is to physically, chemically and radiologically characterise the contents of cell 900.

A 3-D computer model has been created of cells 900 to 905 of the ELAN IIB workshop. Cell 900 is a blind cell with dimensions 7.6 m long, 4.9 m wide and 4.75 m high. The entrance is closed by a barite brick wall. In the cell there are 8 stainless steel tanks. The floor is covered by a stainless steel liner. Cell 900 has been inspected by video through an opening from the adjacent cell 901 in 2002 and the level of radiation was measured by probe (IF194). The measured level was 7 mGy/h.

The current characteristics of the cells will physically measure the position of the tanks, the activity levels of the solutions in the tanks as well their chemical compositions. Based on tenders a mechanical engineering company with nuclear experience has been chosen as the contractor to perform the characterisation.

A first inspection of cell 900 has been carried out using a BROKK machine. It is proposed to build an intervention cell adjacent to cell 900. This intervention cell will have 3 zones:

- A work zone for the operators.
- An intervention zone with cameras.
- A maintenance zone for the equipment used.

In order to simplify the work it is proposed to replace the baryte brick wall with 2 shielded steel doors.

The intervention cell was built in 2004. Studies will be made on the method of collecting samples from the tanks. Samples will be collected after permission from the safety authorities with the results known shortly.

A1.3.7 APM, Marcoule, France

APM (Atelier Pilote de Marcoule) was a pilot reprocessing plant that was used for developing the reprocessing techniques for natural uranium fuels, fast breeder fuels and light water reactor fuels. It consists mainly of:

- Building 214, which has the head end facilities i.e. reception, shearing, dissolution and classification.
- Building 211, used for chemical extraction, separation, purification, R&D laboratories, wastes and effluent storage.
- Building 213, where vitrified fission products were in wet storage.

Building 211 was built between 1960 and 1963 and was used for developing reprocessing processes for natural uranium fuels and later for experimental vitrification of High Level Liquid Waste. Finally, after 1973, it was used for reprocessing fuels from the Rapsodie and Phenix fast reactors.

Building 214 was built between 1980 and 1988 as the head end facility for reprocessing fuel from the Super Phenix reactor. Later, the TOR line was refitted to study the reprocessing of MOX fuels from PWRs.

The APM facility was finally shut down in June 1997 partly due to obsolescence of the facilities as well as for economic reasons. Building 211 will be decommissioned to Stage 3.

Some of the difficulties expected with the decommissioning of the APM are:

- It is a huge, heterogeneous, complex facility with:
  - 760 rooms.
  - 30 high activity cells.
  - 5 shielded production lines.
  - 230 glove boxes.
- The nuclide spectrum is very varied.
- Various radiological incidents have taken place during operation.
- The high activity cells are either “blind” or difficult to enter.
- There is a large quantity of operational waste.

The decommissioning strategy will utilise the lessons learnt during the dismantling of the AT1 reprocessing plant. A “hard decontamination” has been done on the process lines in order to eliminate as much of the nuclear materials as possible. The methods used will be:

- Remote controlled for the high activity cells.
- Manual with long handled tools for the small irradiating beta/gamma cells.
- Manual for the medium and low active cells.

After dismantling of the equipment and the cells are drained and decontaminated, the concrete biological shields will be removed and the steel waste will be sent for melting.

The decommissioning operations are expected to take 10-15 years and cost MEuro 450.

A1.3.8 UP1, Marcoule, France

UP1 (Usine Plutonium 1) was an industrial reprocessing plant designed for GCR, fast breeder and MTR fuels. It consists mainly of 12 buildings:

- Buildings 140, 144, 145, 146, 147 and 148 for reception, storage, shearing or decladding.
- Buildings 100 and 117 for dissolution, chemical extraction, separation, purification and plutonium production.
- Building 98 for uranium storage.
- Buildings 96 and 113 for liquids fission products storage.
- Building 130 for vitrification and wet storage.

The majority of the buildings were built between 1956 and 1963; Building 130 (Atelier de Vitrification Marcoule) between 1972 and 1978; Building 144 (new decladding facility) between 1978 and 1983. During 40 years, most of the equipment was refitted.

UP1 was finally shutdown in December 1997 after treatment of all the French and Spanish GCR fuels (more than 18 000 tons including all kinds of fuels during 40 years of operation).

UP1 will be decommissioned to Stage 2.

Main UP1 features are:

- Large, heterogeneous complex facility with:
  - 600 restricted rooms and cells.
  - 1 700 m^3 of vessels.
  - 15 000 m^3 of pounds.
  - 5 000 tons of process equipment.
  - 20 000 tons of structural material.
  - 6 shielded production lines.
  - 145 glove boxes.
- Very varied nuclear spectrum.
- Various radiological incidents during operation.
- Large quantity of operational waste.

The decommissioning strategy will utilise the lessons learnt during 40 years of operation. Conventional and specific rinsing of the process lines have been achieved in order to eliminate as much of the nuclear materials as possible. Then, the methods to be used will be:

- Remotely controlled for the high activity cells.
- Manual operation with long handled tools for the small irradiating beta/gamma cells.
- Manual operation for the medium and low active cells.

After dismantling the equipment, the cells are drained and decontaminated to less than 100 Bq/cm^2.

The decommissioning operations are expected to take 25-30 years and cost MEuro 2 000 (MEuro 3 000 including support facilities: laboratories, liquids and solids waste treatment).

A1.3.9 Wiederaufarbeitungsanlage Karlsruhe (WAK), Germany

The Wiederaufarbeitsanlage Karlsruhe (WAK) was a pilot reprocessing facility located on the grounds of the research centre of Karlsruhe (FZK). It was shut down in 1990, for political reasons.

The plant had a design throughput of 35 t of spent fuel/year at a maximum burn-up of 20 000 MWd/tU. The plant was put in hot operation in 1971. During its operation, it processed 200 t of heavy metal, including 1.8 t of plutonium. Because of test operation with fuel at a burn up up to 40 000 MWd/tU, the average-burn-up was as high as 26 000 MWd/tU.

The owner of the plant is the Federal Government of Germany. The operator is the WAK BGmbH, until 1980 a subsidiary of the chemistry industry and from 1980, a subsidiary of the nuclear power station operating utilities.

For decommissioning and dismantling the facility, a letter of understanding was signed between the Federal Government and the utilities. Under this agreement a contract was established between the FZK as the responsible Project Manager and WAK BGmbH as the executing company.

As the WAK project was expected to utilise remote dismantling to a considerable degree, a remote dismantling test facility was built in part of the turbine hall of the MZFR after that plant was dismantled. The objective was to demonstrate and verify difficult remote dismantling steps (before performing them in active conditions), partly to train the crews to perform effectively and partly in support of licensing activities.

A full scale mock-up of the WAK Cell VI (for medium active liquid waste) was erected inside. For simulating the dismantling operations, the following equipment was installed:

- A manipulator carrier system.
- A replica of the cell hall crane.
- 2 electro-mechanical master-slave manipulators.

The operations were remote controlled and monitored from a control room with 3 work stations, through 32 TV cameras, 42 TV monitors and 11 control consoles. The facility was operated by a team of 11 remote operators and 4 engineers for planning/preparation of waste.

Dismantling the WAK started in 1993. It is being performed in a series of six steps, the first two which have been completed. These were the decommissioning of the process building and the early dismantling in the process building. In Step 3, the aim is to “free” all controlled areas in the process building.

Step 3 was carried out under a series of licences and the work under the licences already granted includes:

- Vertical and horizontal dismantling in the process cells.
- Dismantling of the dissolver.
- Semi-remote dismantling of the pipe duct with the mixer settlers.
- Manual dismantling of laboratories and auxiliary systems.

Some detailed planning has been affected by the experience gained. The cell containing the uranium and plutonium final product vessels was manually dismantled. Another cell, with the high active feed vessels, originally planned to be remotely dismantled horizontally, will now be dismantled vertically.

The dual armed manipulator used in the remote handling workshop in the MZFR turbine hall has now been installed in the crane hall, with the control room at one end. New remotely operated equipment for vertical dismantling has been installed in the hall above the cells and successfully tested. Three-shift operation has been implemented and trained in cold operation. The first of four large process cells has been emptied.

Horizontal remote dismantling has been used to empty one process cell. Semi-remote dismantling with long handled tools from a shielded carriage was used when the pipe duct including the mixer settlers was emptied, except for two medium active waste buffer vessels. Manual dismantling with workers wearing masks of the valve gallery and chemical supplies installation has been completed.

Large components are cut into pieces with a maximum length of 2.5 m for transport reasons. Cut pieces are put into troughs and taken to cells for further segmenting, before being loaded into drums with a double lid system. Step 3 was completed by June 2005. Between 2004 and 2005, the plant was constructed (by FZK) for vitrification of the high active waste concentrates.

An interesting comparison could be made between the manual dismantling of cell VII and the vertical remote dismantling of cell V. In cell VII, in 4 800 man-hours dismantled 17 t of material with a total activity of 5×10^{11} Bq. The maximum dose rate in the cell was 0.02 mSv/h and the collective dose to the workers was 8 mSv. In the remotely dismantled cell VI, 18 000 man-hours produced 33 t of material with 3×10^{11} Bq. Here the maximum dose rate was 120 mSv/h and the collective dose was 6 mSv. The manual operation was carried out with a one 8-hour shift a day, while the remote operation was round the clock with three 8 h shifts.

A1.3.10 JAERI's Reprocessing Test Facility (JRTF), Japan

JAERI's Reprocessing Test Facility (JRTF) was constructed during 1959-68 and operated for two years before shutdown in 1970. The Purex process was used to recover about 200 g of plutonium.

The facility consists of a main building with the reprocessing plant and two annex buildings for storage of the liquid wastes. The annexes are connected to the main building by ducts. The main building has a floor area of about 3 000 m^2 and the annexes have 160 m^2 and 400 m^2 floor areas respectively.

The project to decommission the JRTF was started in 1990. The project has three phases. During the ongoing Phase 1, the liquid waste arising from the operation is being conditioned. Phase 2 is the research and development work of decommissioning technologies for dismantling the JRTF. Ongoing Phase 3 is the actual dismantling activities.

The liquid waste consists of:

- Alpha-contaminated liquid waste.
- Spent solvent.
- Unpurified uranium solution.
- High-level liquid waste.

Treatment of the 60 m^3 of alpha-contaminated liquid waste was completed by the end of 1995. The treatment of the spent solvent was by washing first to remove plutonium, to incinerate the washed solvent and to condition the ash in cement. This treatment was completed by the end of 1995.

The plutonium in the unpurified uranium solution was adsorbed by inorganic adsorbents, then the solution was solidified. The caesium, strontium and plutonium in the high-level liquid waste were also taken up on suitable inorganic adsorbers. The treatment of all liquid wastes was completed in 1998.

R&D for the dismantling of the JRTF was started in 1993.

The research and development programme resulted in the following:

- A three-dimensional CAD system was developed for the dismantling procedures.

- A robot carrying a TV camera and distance measurement device was constructed for acquiring data in high radiation areas.
- A remote dismantling (and segmenting) machine has been designed and built for large tanks.
- Concrete decontamination up to a depth of 10 mm by laser techniques has been developed.
- Improved protective suits were developed for working in alpha-contaminated areas.

Actual dismantling activities started in the main building of the JRTF early in November 1996. To support the dismantling activities, the surface contamination was nuclide specifically determined on sample pipes cut-out from various typical sections.

The glove boxes were dismantled in 1996 to prepare the space for temporary waste storage yard. Then, the analytical cells which consists of an inner box of stainless steel with iron and lead shielding, components in hot cave, solvent recovery cell, Pu cell and large sized tanks were dismantled. As a result, about 45% of components in controlled area have been dismantled. Workers wear ventilated suits to prevent internal exposure. The dismantling activities are carried out with three shifts.

A1.3.11 Plutonium Fuel Fabrication Facility, Japan

Japan Nuclear Cycle Development Institute has 3 facilities for the development of plutonium fuel:

- Plutonium Fuel Development Facility (PFDF) started in 1966 for basic research on plutonium and MOX fuel including manufacturing of irradiation test fuels.
- Plutonium Fuel Fabrication Facility (PFFF) had operated from 1972-2002 for fabricating of MOX fuels for ATR Fugen and experimental FBR Joyo.
- Plutonium Fuel Production Facility (PFPF) started in 1988 to produce MOX fuels industrially.

The D&D project for PFFF is divided into the following four phases, basic concept of each phase is discussed:

- Phase 1 (up to 2010): stabilisation and shipment of nuclear material in the facility. Decontamination and volume reduction techniques will be chosen.
- Phase 2 (2010-2015): D&D planning and adaptability tests.
- Phase 3 (2015-2020): size reduction of equipment and GB. Promotion of R&D.
- Phase 4 (2020-2035): re-use of buildings for waste storage.

Glove box dismantling equipment, that has a size reduction area with an arm type robot and manipulators, has been operated in PFPF. The purpose of this equipment is to dismantle used glove boxes from the MOX pellet fabrication process in PFPF by an ordinary method (by workers using personal protection equipment (ventilated suits) and a remote controlled method. The plasma arc cutting system and mechanical tools (abrasive disc, chip saw, rotary band saw and nibbler) are used for the dismantling of the GB body and equipment. Various data for size reduction has been collected. The cost reduction of GB dismantling work by remote operation is to be expected. This data and knowledge will be reflected in the planning of the D&D project for PFFF.

A1.3.12 Uranium Conversion Plant, Korea

The Uranium Conversion Plant of KAERI, with a capacity of converting 100 t/year of yellow cake to UO_2, started operation in 1982. In 1987, an in-house developed process, the ammonium uranyl corbonate or AUC process, was introduced and used for producing 320 t UO_2 for fuel for the Wolsong-1 CANDU reactor. The plant was shut down in 1992, as cheaper UO_2 could be bought on the international market.

The project to decommission the plant was started in the year 2000. Two of the main actions in the decommissioning programme are to decontaminate stainless steel equipment to reduce radwaste and to decontaminate the building walls to levels for unrestricted re-use. After decontamination, the equipment and the walls have to be checked for alpha contamination. However, alpha spectroscopy is time and effort consuming and difficult to carry out practically. So gamma spectrometry has been proposed to be used at the conversion plant as a uranium measurement tool. Experiments have been performed to test the feasibility of this approach. Specifically the experimental aims were to use gamma spectroscopy:

- To show that metal components could be decontaminated to release (background) levels.
- To show that the inside walls of the conversion plant could be decontaminated by removal of surface concrete.

For the second point, concrete samples were taken at the surface, 10 mm depth and 50 mm depth, crushed and homogenised. Gamma measurements were made directly, alpha activities by electro-deposition of material recovered from the elute solution.

Some of the results are summarised below:

- Measured gamma radiation intensities for trace amounts of AUC were compared with calculated values. Two types of detectors were used:
  - low energy photon detector;
  - coaxial photon detector.

  Both with a detection limit of about 0.01 Bq/g. The measured values matched the calculated one. In the low energy region, measured values were higher than those calculated.
- In the alpha and gamma spectrometry tests on metallic samples and concrete powders, they matched each other well in the activity region above 0.01 Bq/g. Below this level, gamma intensities were higher than the alpha. This will lead to a slight overestimation by using only gamma spectrometry.

From the above, it could be concluded that gamma spectrometry could be used for the release measurements at the Uranium Conversion Plant.

A1.3.13 Active Chemical Laboratory (ACL), Sweden

The Active Chemical Laboratory at Studsvik, Sweden, and its associated ventilation and filter building are currently in a Stage 1 status. The aim of the project is to bring it to a Stage 3 condition, the main incentive being to avoid the high costs of heating and ventilation of the large building with a total floor area of 12 430 m^2 (with an additional 1 600 m^2 in the filter building).

The ACL has, apart from water and heating system, also systems for de-ionised water, steam, pressurised gas, air, etc. The effluents are separated into three categories and piped underground to a separate treatment facility before release into the Baltic Sea. The building has three separate filter systems, two (from the glove boxes and the cells) with pre-filters and the third (for general ventilation) directly connected to the main filter bank.

A pre-project had been conducted in 1998 to make a first radiological characterisation and, based on this, produce a decommissioning plan, with time schedule and estimated costs. The pre-project also included the remediation of the asbestos insulation on the site. The study indicated a cost of MSEK 16.8 over a three-year period. The period was reduced by the owners later to 2 years.

The project group and the authorities agreed on project specific surface and volumetric activity concentrations release limits and extent of measurements with smear tests, scintillation instruments and ISOCS. The project was divided into areas, area 1 being the loft of the ACL building.

When project work got started, with area 1 as the starting area, it was soon realised that the work involved in the measurements was much more time and effort consuming than had been foreseen. It was decided to carry out a 2-month planning phase (in August/September 2000) in order to produce a more realistic time schedule and estimated costs.

This planning phase was executed with reduced personnel and on the basis of experiences acquired from area 1 as its basis. It resulted in a total cost for dismantling, decontamination and free release of the buildings of MSEK 65. Waste measurements and collection are included in these figures but not processing, storage and disposal. The time schedule for the project has been expanded from three to seven years. The re-planning of the project has been performed by a group of five people during two months.

Canadian AECL projects have had similar experiences regarding time and cost estimations.

In the ACL project, ISOCS is used for scanning of larger areas (~ 4 m^2). The aim is to detect gamma radiation and by the use of correlation factors make an estimate on alpha contents. Measuring time can be tens of hours. Smear tests are used on small areas.

A1.3.14 BNFL 204 Primary Separation Plant, United Kingdom

The B204 building was originally built to reprocess uranium metal fuel and operated from 1952 until 1964 when the plant was superseded. One of the two process lines was converted to reprocess oxide fuel, operating from 1969 to 1973 when a release of activity into operating areas permanently stopped operations. It is now being decommissioned to a Stage 2 status.

The building is of a reinforced concrete core about 60 m in height surmounted by a 60 m ventilation stack. The two original mirror image process lines each comprise two highly active cells and a medium active cell. The decommissioning strategy has divided the 20-year project into nine phases with safety and financial sanction sought separately for each phase.

The first three phases have been completed:

- Phase 1

  The construction of a building for waste handling facility (WHF) and provision of medium activity cell north (MAN) decommissioning equipment. The site clearance for the WHF was completed in August 1992 at 81% of estimated cost and 30% of estimated dose uptake. After design and tendering in 16 work packages, the building and civil engineering work has been completed and all mechanical equipment has been procured and installed. The plant for decommissioning the MAN cell has been inactively commissioned and is awaiting agreement from the site regulators to commence active work.

- Phase 2

  Design studies for remaining project phases and development of project remote handling technology based on the CODRO concept (Contact Development Remote Operation).

  Conceptual design studies for high active and remaining medium active cells have been completed. Before finalising the strategy for the high active cell remote decommissioning, operational data from the MAN cell will be evaluated.

- Phase 3

  Provision of a new filtered cell ventilation system was required before decommissioning operations could commence within active cells. This phase of the project is complete at a cost of £1.5 million.

Work is proceeding on the other phases:

- Phase 4

  Decommissioning the MAN cell started with the removal of the associated control systems and out-cell services. The MAN cell roof is partly under the high active north outer (HANO) cell. Due to the risk of the collapse of supporting equipment in the HANO cell, work in the MAN cell was suspended and the personnel were redeployed for clearing outcell vessels and piping.

- Phase 5

  Emptying the stainless-steel hulls silo. This has been advanced by ten years as a result of the Phase-2 design studies, which showed that completion by December 1996 would match an availability "window" at the waste disposal facility B38. A flask loading facility has been installed, with a remotely operated loading vehicle (ROV). The ambient dose rates in the silo are about 75 mSv/h gamma. The B38 facility which receives the flasks with the removed hulls was due to be closed in November 1999. The Remotely Operated Vehicle (ROV) used in this operation showed reliability problems. So a second vehicle was purchased and seven days a week working was adopted to meet the schedule.

- Phase 6

  Decommissioning of the Medium Active South (MAS) cell. Planning and purchase activities have been going on.

- Phase 7 and 8

  Decommissioning of the high active cells north and south outer (HANO/HASO). HANO has been used earlier as a venting cell, with filter banks in the upper part of the cell. Condensing nitric acid vapours have, over the operational period, corroded the cell vessels. Video investigation (part of which was shown to the TAG) revealed damage of some of the supporting structures and a vessel that has fallen to the bottom of the cell. The most urgent problem is that the criticality safety cases are built on the assumption that the cells and structures are in a stable condition. So a new criticality warning system has been installed and components are being removed from the cell.

  Like the UKAEA, the UK Group of BNF plc has been reorganised. Later BNF (Inc) had purchased Westinghouse (100% of the electrical part and 60% of the Government and Environmental Departments). This has led to further reorganisation.

A1.4 Completed fuel facility projects

A1.4.1 Tunney's Pasture Facility, Canada

The Tunney's Pasture Facility in central Ottawa was used for research, production and worldwide shipping of radioisotopes. After thirty years of operation, it was shut down in 1984. A first decommissioning phase was carried out to reduce the licensing to a possession only level. This phase was completed in 1987. Planning for decommissioning for unrestricted release was started in 1989. The authorisation for starting work on site was received in 1991. This second phase was completed in August 1993.

The total activity inventory in the plant was estimated to be less than 1.48×10^{10} Bq (4 Ci) including a number of difficult to measure nuclides like ^{63}Ni. The radioactivity was mainly located in the ventilation system, which was dismantled first. This was technically not demanding but required nine months of fully suited work and rigorous personnel discipline.

The major engineering work was in connection with the removal of the eight hot cells, typically with 1-m thick walls of heavy concrete, clad with 13 mm carbon steel and 4 mm stainless-steel linings. All contaminated components were first removed. The cells were then cut up using diamond wire saws.

The high background dose due to the accumulation of radwaste from the decommissioning hampered the progress of the project. The radwaste continued to accumulate because of the need to characterise the waste in a manner acceptable to the organisation taking the future responsibility for it, AECL Research. Agreement was finally negotiated between the project and AECL Research and the project rapidly progressed towards completion.

The project employed 30 persons at its peak, divided into three groups: decommissioning, radioprotection and health physics.

The final survey of the site was carried out by project teams, after which the Atomic Energy Control Board (AECB) audited the building and the survey records.

The de-licensing is based on a release level of an average ambient dose rate of 13 µR/h (total, i.e. 5 µR/h above the average background of 8 µR/h), with an assumed occupancy of the premises of 2 000 h/a. This would give a maximum individual dose of 260 µSv/a. The inclusion of the contribution from naturally occurring radioactivity makes it difficult, however, to make comparisons with other recommendations in connection with free release levels, most of which exclude the naturally occurring component. The facility was formally released by the AECB in January 1994.

A1.4.2 BNFL Co-precipitation Plant, United Kingdom

The BNFL Co-precipitation plant was part of the fuel reprocessing operations at the Sellafield site and produced a mixed powder of plutonium dioxide and uranium dioxide for the first fuel charge for the Dounreay PFR. The plant was in operation between 1969 and 1976. This Stage 3 decommissioning project was run as a pilot project, for acquiring data on the decommissioning of fuel facilities.

The first decommissioning activities were the post operational clean out and the dismantling of the wet chemistry suite. Some 130 l of flushing liquor containing 900 g of plutonium and 1 400 g of uranium were reprocessed.

The next main operation was the removal of the Ball Mill from its glove box containment. This was the first application of the reusable modular containment (RMC) and demonstrated many of its advantages over a PVC-tent arrangement.

Subsequent removals included the powder transfer equipment as well as the furnace suite. For the latter, a flexible PVC enclosure was used instead of the RMC, thus allowing a comparison of the two procedures. In addition, the geometrically safe plutonium and uranium nitrate storage tanks were dismantled and the removal of the remaining glove boxes completed in October 1990.

Strippable coatings were used as a protective pre-coat on RMC panels before radioactive work or as a tie-down coat to fix loose activity. This also simplified the final clean up. Small-bore pipe-work was dealt with without loss of containment by means of crimping/shearing tools.

The project was originally scheduled to be completed by late 1988. It was actually completed in March 1991. The main reason for the delay was the prolongation of the R&D activities to maximise the project's usefulness as a pilot project. There were however some other reasons as well such as:

- Greater than expected fissile material left in the systems.
- Differences between drawings and actual plant.
- Priority given to production operations over decommissioning projects at shared facilities (e.g. pressurised suit entry facility).
- Unplanned maintenance work.

The final cost is anticipated to be £2 245 000 compared to the originally estimated £2 033 000 (both 1989 values) excluding TRU treatment and disposal costs. The increase in anticipated final cost of £212 000 is attributable to labour costs for the extra dismantling required offset by some savings on Plant and Equipment.

Other project data of interest:

- Person-hours 19 730
- Collective dose 305 milliman-Sv
- Waste:
  - Plutonium-contaminated material (PCM) 44.4 m^3
  - Shallow-land burial 12.0 m^3
- MOX recovered 46.1kg (+2.9 kg as nitrate)

A1.5 Former CPD projects which are no longer participating in the Programme

A1.5.1 Kernkraftwerk Lingen (KWL), Germany (This project is no longer a participant in the CPD Programme.)

KWL Lingen was an indirect cycle 520 MWt boiling-water reactor with oil-fired superheater that operated from 1968 to 1977. It was placed in a Stage 1 "Safe Enclosure" (SE) status in 1988. The licence is valid for 25 years. The conditions of the safe enclosure status are essentially:

- The safe enclosure consists of the reactor building, the waste treatment building and the building interconnecting them.
- All pipes penetrating the safe enclosure were cut and sealed, systems remain open.
- All openings from the safe enclosure are closed, shut and sealed, except for one door.
- All liquids have been drained from the systems.
- A small air conditioning plant has been installed to keep the air below 50% relative humidity.
- A small exhaust system has been installed for the controlled release of air, which is filtered and monitored.

The operation of the dormancy of the plant has been without any incidents. The leakage of air and activity release from the safe enclosure is being monitored under a co-operative programme with Euratom. The only aspect of interest noted was that the relative humidity in the enclosed area was higher than expected. The reason was identified to be the design of the drying system. A new drying system had been taken into operation in 1994.

The plant is inspected periodically. The systems in operation are inspected according to an inspection programme prescribed in an operations handbook. The costs for the operation are under DM 1 million per year.

In 1996, the company decided to try to utilise the availability of a volume quota at the Morsleben repository for sending the operational waste at the station. This consisted of ion exchange resin, bituminised evaporate concentrates, filters and miscellaneous waste. A licence to establish a suitable infrastructure for this was obtained in November 1997, by which the scope of a number of service systems were extended, including the ventilation, condensate and electric systems. The allowed discharge volume to the river was increased from 500 m^3/a to 5 000 m^3/a. Changing facilities, rest rooms were to be extended and elevators and lifting devices to be renovated.

However, the Morsleben site was shutdown in September 1998 and the planned waste disposal could not be realised. Meanwhile a decision to treat all the waste stored in the reactor building has been taken. Furthermore a license was applied for and granted. All kind of wastes are under treatment except for the ion resins which will remain on site. Besides this work, a cost study and a risk evaluation for a possible change of strategy from the actual safe enclosure status to a full decommissioning status is in preparation.

A1.5.2 Shippingport, United States (This project is no longer a participant in the CPD Programme)

The Shippingport Atomic Power Station was constructed during the mid-1950s under the President Eisenhower's "Atoms for Peace" Programme. The station achieved criticality in December 1957 and was operated by a public utility, Duquesne Power and Light Company, under supervision of the United States Atomic Energy Commission and later the Department of Energy-Naval Reactors Programme until operations were terminated in October 1982. The station's nominal power output was 72 MWe. Over the operating life of the station there were 2 246.8 effective full-power days and the total gross generation was 7 374 GWh.

The objectives of the decommissioning project were to:

- Demonstrate the safe and cost effective dismantling of a full-scale nuclear power plant.
- Transfer the experience of such a project to the nuclear industry by using a large number of sub-contractors.
- Document these experiences in detail for use in future decommissioning projects.

Conceptual and detailed engineering for the decommissioning project was completed in 1983. The plan was to decommission the plant to Stage 3. The physical decommissioning consisted of the demolition and disposal of 26 various fluid and electrical systems before the buildings could be demolished. In all, about 17 100 m of contaminated piping and 16 800 m of non-contaminated piping, and 1 300 tanks were removed. All buildings were demolished and removed to about 1 m below the ground level. The Reactor Pressure Vessel/Neutron Shield Tank assembly, which measured 12.5 m high by 5.4 m in diameter, transported by barge 13 525 km in 44 days to the burial site. The total radioactivity removed was 6.14×10^{14} Bq, of which 6.09×10^{14} were contained in the reactor vessel. Total project radioactive waste volume disposed of was 6 057 m^3 weighing approximately 4 185 t. Also 11 470 m^3 of non-contaminated rubble was created during building demolition and was used to backfill the below grade reactor building enclosures.

Physical work on decommissioning the Shippingport reactor started on site in September 1985. All physical decommissioning work on site was completed on July 1989, about six months ahead of schedule. The total project cost was US$91.3 million, US$7 million less than the estimated US$98.3 million. The approval for release of site was issued in December 1989.

The most significant part of the Shippingport project was the one-piece removal of the reactor pressure vessel (RPV) package and its 8 400-mile shipment. The total cost to prepare, to remove, and to bury the package was US$10.3 million. Work included in this was: re-positioning the non-fuel reactor internal components in the RPV; filling the RPV cavity and the NST annulus with an engineered grout mixture; developing and writing a Safety Analysis Report for Packaging; removing the RPV package as a single package, then loading and transporting the package to the DOE Hanford disposal site for burial, and the required co-ordination of shipment and state notification activities.

The total personal exposure was 1.55 man-Sv to be compared with an estimated 10 man-Sv in the original decommissioning plan.

The main lessons learnt from the project were:

- One-piece removal of the reactor vessel was cost effective and practical. However, it is worthy of note that the low radiation levels of the plant and the low burial costs at the government owned burial ground were advantages which may not apply to the decommissioning of large commercial plants.
- Existing technology and equipment can accomplish decommissioning of nuclear power plants at reasonable costs.
- Observation of ALARA practice coupled with careful planning and scheduling can reduce radiation exposure and raise productivity levels.

A1.5.3 Experimental Boiling-water Reactor (EBWR), United States (This project is no longer a participant in the CPD Programme)

The experimental boiling-water reactor (EBWR) at the Argonne National Laboratory was a demonstration BWR, originally of 20 MWt (5 MWe), then upgraded to 100 MWt. It started operation in 1956 and was shut down finally in 1967.

The first phase of the project – preparatory work for decommissioning – was completed in 1988. The removal of the primary and secondary system components, which constituted the second phase of the project, was completed in 1989. During the third phase of the project, which covers the removal of the reactor vessel and internals, there have been a number of major changes in the schedule and the operations of the project:

- All Argonne site construction activities were shut down between November 1990 and February 1991, ordered by USDOE internal inspection team. The Argonne engineering group provided some support to the decommissioning programme and so this impacted the project greatly even though the project was not criticised by the team. Even after the re-start of activities, planning and implementation of additional management oversight and quality assurance provisions significantly affected the progress of the project.
- The project started as one executed by an in-house skilled Argonne work team, consisting of three to 10 persons. Due to the limited availability of skilled decommissioning technicians, the project approach was shifted from using an in-house team to using an external fixed price contractor: the Alaron Corporation.
- Originally, it had been planned to use abrasive water jets to segment the entire reactor vessel, mainly to reduce the fire risks of using "hot" cutting methods. Due to the change of scope at the placing of the Alaron contract and the new time schedule, the vessel was segmented using a WACHS cutting machine which uses a "cold" mechanical cutting technique. Fifty linear feet of the vessel was cut using the abrasive water jet technique. The mechanical milling machine worked very well and this allowed the comparison of the two techniques.
- Two of the four lifting slings broke when transferring the core structural assembly out of the reactor vessel, due to an unobserved protruding lug on the outside of the core shroud fastening in the vessel opening. There were no serious consequences.

The reactor core assembly was transferred to the fuel pool in one piece. It was size reduced for disposal using an under water plasma torch. This technique was used sparingly inside the reactor vessel because of the redwood (sequoia sempervirens) liner behind the vessel wall.

All vessel wall pipe penetrations were cut using a WACHS split frame pipe cutter. The WACHS split frame inside diameter cutting machine was then used for horizontal cuts, first to separate and remove the vessel bowl and then to divide the barrel of the vessel into five rings. The rings were lifted out from the vessel cavity and size reduced in a cutting tent. An abrasive water jet was used to perform a test cut 15 m long.

Comparison of the three cutting methods used on the EBWR reactor vessel (plasma torch, abrasive water jet, WACHS mechanical cutting machine) showed that, in this particular application, the WACHS machine had the most advantages. A comparison was also made of all the cutting methods used in the project as a whole.

Part of the bio-shield behind the reactor cavity liner consisted of lead bricks, more than half of which will be recycled (by melting) by a nuclear research facility for use as shielding. The activated concrete was removed using a BROKK machine.

The EBWR facility was converted for use as an interim storage facility for transuranic waste. The project was initiated in April 1986 and completed in February 1996. The total costs were US$ 19 586 Million.

A1.5.4 Fort St. Vrain, United States (This project is no longer a participant in the CPD Programme)

Fort St. Vrain was a 350 MWe high-temperature gas-cooled reactor that was operated by the Public Service Company of Colorado (PSC) between 1976 and 1989. It was shut down mainly due to the poor operational performance (<15% capacity factor/<30% availability), high fuel costs and consequently uneconomic to operate.

Originally the spent fuel was to be stored or reprocessed at the Idaho National Laboratories (INEL). As INEL refused to accept the fuel, PSC constructed an intermediate dry storage facility for the fuel on site with a 20-year licence (+20 years option).

Immediate dismantling to Stage 3 was chosen as the decommissioning alternative for a number of reasons, including:

- Increasing disposal costs with time (11.9% per year since 1980).
- Uncertain long-term regulatory situation.
- Adequate dismantling technology available.
- Technical personnel with intimate knowledge of site would not be available later.
- Easier to "repower" the site with a gas fired boiler.

The Westinghouse Team with M.K. Ferguson as construction contractor won the fixed price contract for decommissioning the plant. The total costs, including in-house costs and that for low-level waste disposal were estimated to be USD$174 million. The dry fuel storage costs were US$13 million.

One characteristic of the Westinghouse concept was to dismantle the reactor internals after filling the vessel with water. This was done by a 325 000 gallon water system, with two pumps and ion-exchange and 0.3-5 μm filters for keeping the water clear.

First, the central part of the top slab of the prestressed concrete vessel was cut out in 12 wedges using a diamond wire saw, involving the removal of 1 320 t of concrete. A rotary work platform was installed for the continued work. The top head liner was cut up with oxygen lances and removed, after which the reactor internals were removed with the water in the vessel acting as shielding. These activities included the following:

- The graphite from the reactor vessel – 1 770 pieces with surface dose rates up to 3 Sv/h – was sent off site as low-level waste.
- The upper core barrel (9.15-m diameter, 8.85-m height, 67-mm wall thickness) was segmented under water with a remotely operated plasma arc torch. The segments were shipped to Hanford as low-level waste.
- Two shifts of eight divers, each diver making a 90-minute dive, were utilised:
  - First to clean up debris, etc., from the core support floor.
  - To use underwater jack hammers to remove silica plugs in the core support posts.

- To use a remote plasma arc cutting tool to free inconel sleeves in the posts.
- To use handheld plasma arc torches to remove the stainless steel floor thermal seal.

- The inconel sleeves had contact doses of 200-500 mSv/h and so steel work platforms were designed to keep the divers at a safe distance.
- The core support floor was cut loose from its supports during 1 250 dives, performed over ten months. A collective dose of 173 man mSv was taken.

A 4-inch steel plate had been attached to the top of the core support floor as shielding. This reduced the dose rate from the floor (when lifted out) from 10 mSv/h to 0.5-0.6 mSv/h.

The entire facility was cleaned up to release limits of 25% of the guideline value of 5 μR/h. The final radiation surveys and the decommissioning project were completed in early 1997. The USDOE accepted title to the fuel and agreed to pay the power company the costs for the dry storage facility for fuel on site.

A1.5.5 Paldiski, Estonia (This project is no longer a participant in the CPD Programme.)

A Soviet submarine training centre was operated by the USSR Navy at Paldiski, Estonia, from 1968 to 1989. The centre comprised a number of nuclear facilities, including two submarine hulls, each containing a nuclear reactor. After shut down in 1989, the reactors were defuelled and the fuel transported to Russia in 1994. Certain non-contaminated and secret equipment was also dismantled and removed off site.

Both reactors are housed in a common building, with other buildings on the site for relevant auxiliary facilities such as treatment and storage of liquid and solid wastes, laboratories, as well as ventilation, heating and laundry facilities. The reactors, each with its primary system, have been enclosed in separate concrete sarcophagi. In addition to the external shells, the sarcophagus also includes concrete placed internally as intrusion protection.

Ownership of the site was transferred to Estonia in late September 1995. In order to both advise and help the Estonian authorities to proceed with a safe and timely decommissioning of these installations, an international expert group named Paldiski International Expert Reference Group (PIERG) was created in 1994. This group drew up a Conceptual Decommissioning Plan, which constitutes the basis for on-going decommissioning work and planning of future site activities.

A company, ALARA Inc., was established and is funded, by the Estonian government, for the purpose of managing the Paldiski site and associated decommissioning activities as well as being responsible for all radioactive waste within Estonia.

The project is characterised by some unique features:

- It is a nuclear decommissioning in a country without a nuclear programme. So there is no existing infrastructure such as a qualified nuclear technology industry or power plant health physics operators.
- With no accumulated decommissioning funds and limited available state funding, the work to be carried out must be very severely prioritised.

The decommissioning project aims at establishing a waste management system with a long-term monitored interim storage and minimising the extent of the controlled area. For this, a number of operations are going on, such as:

- An interim storage has been constructed in the Main Technological Building. It can take 720 standard sized (1.2×1.2×1.2m) waste containers in two cells and has been in operation since 1997.
- A waste receiving and treatment area is being established in an annex to the interim storage.
- The solid waste storage has been cleaned out. This was one of the most complicated and work-intensive projects. The building consisted of 10 cells with 50 cm thick wall; with additional brick walls and earth fill as shielding for the outer walls. Only three of the cells contained waste, but this consisted of very mixed material from control rods to steam generators, circulation pumps, pipes, wood, plastic sheet, rags, filters, etc. These had been dumped into the cells without any conditioning, segregation or packaging. The waste was characterised by an international team: the Estonian ALARA AS to co-ordinate, the Swedish Studsvik RadWaste AB and SKB for dose rate, nuclide specific and contamination measurements and the USDOE (Idaho Operations Office) for gamma imaging with the GammaCam technology. The retrieval work has resulted in 76 concrete waste container of 7 m^3 each, 3 control rod containers, 8 steam generators, and 67×200 l drums with compacted waste.
- A European Commission PHARE financed project has been completed on a feasibility study for dismantling the Liquid Waste Treatment Facility (performed by SKB, Sweden and SGN, France). The study recommends:
  - Immediate dismantling (not deferred).
  - Manual dismantling (rather than remote).
  - Use of in-house ALARA AS personnel as far as possible.

The total cost for the dismantling is estimated to be 16 MEEK, without waste disposal. The disposal of the 350 m^3 waste can cost as much as another 70 to 140 MEEK, according to present estimates. The disposal costs are so high because of the relatively small quantities to be disposed.

A1.5.6 West Valley, United States of America (This project is no longer a participant in the CPD Programme.)

The 200-acre West Valley Demonstration Project (WVDP) is part of the Department of Energy’s nation-wide environmental restoration and waste management effort. The Project is located at the site of the only commercial nuclear fuel reprocessing facility to have operated in the United States, near West Valley, N.Y., about 35 miles south of Buffalo. The site’s former operator generated more than 600 000 gallons of liquid high-level radioactive waste (HLW). In accordance with the West Valley Demonstration Project Act of 1980, and in partnership with the New York State Energy Research and Development Authority (NYSERDA), DOE’s primary mission at the site is to safely solidify that waste into a durable, solid borosilicate glass – a process known as vitrification – and to clean up and close the facilities used. The vitrified waste is encased in stainless steel canisters and is to be transported to a federal repository for permanent disposal at a later date.

Some of the many accomplishments at the site since its 1982 inception include the processing of over 1.7 million gallons of liquid to produce approximately 20 000 drums of cemented low-level waste; completing transfer of acidic and other wastes to the main waste tank; and constructing vitrification, off-gas treatment, canister load-in, and other facilities. The vitrification facility began radioactive operations in June 1996 and, as of September 2001, has produced 262 canisters of vitrified high-level waste. The primary vitrification campaign was completed in June 1998, and vitrification of the remaining tank heel material is nearing completion. The canisters are currently being stored on-site in a shielded interim storage cell within the main process building.

Progress in other areas is continuing, including construction of a remote-handled waste facility, cleanup of the main plant head-end cells, and shipment of low-level waste off-site for disposal. In addition, off-site shipment by rail of the remaining 125 spent nuclear fuel assemblies was made Fall 2001.

DOE is also focusing on transition from vitrification operations to decontamination and decommissioning activities. This will include preparation of two Environmental Impact Statements (EIS) – a Decontamination and Waste Management EIS for near-term activities and a Decommissioning and Long-Term Stewardship EIS.

A1.5.7 Fernald Environmental Management Project, United States of America (This project is no longer a participant in the CPD Programme.)

The Fernald Environmental Management Project (FEMP) covers the decommissioning and environmental remediation at the Fernald site, where uranium metal used in weapon grade material had been manufactured in some 200 facilities. The main contractor on site is Fluor Daniel Fernald.

The project is divided into five "Operable Units", of which OU3, the facilities closure and demolition project, is the part of greatest interest to the Co-operative Programme. To date the safe shutdown of all facilities and removal of nuclear material have been completed. In addition, 90 of the 273 structures have been dismantled. The original time schedule for the whole project had planned on completion for the year 2008. A request for proposals (RFP) has been issued inviting bids for an accelerated clean up, with significant incentives for earlier completion and penalties for completion later than target date.

The main aim of the safe shutdown activities was the recovery of nuclear material "held up" in systems. In the buildings and complexes that were covered, 690 050 lbs (about 314 000 kg) of nuclear material were recovered. This material was loaded into 55-gallon (200 l) drums and will be sent for disposal at the DOE Nevada Test Site. One of the important lessons learnt during the safe shutdown activities was the use of non-destructive assay (NDA) methods for the location of hold-up material in tanks, piping and process equipment.

Another important on-going activity is the Silos project, where 4 silos (one of which is empty) containing radium and thorium bearing residues are to be emptied and treated. Earlier, Silos 1 and 2 had been surrounded by soil reinforcement, first in 1964 and further upgraded in 1983. In 1991, they had been capped by a one-foot thick layer of bentonite clay to control the radon gas inventory and emissions to the atmosphere. Now various vitrification and other techniques are being studied and tested for treating the residues and metal oxides in order to allow disposal later on at the Nevada Test Site or a commercial disposal facility.

The FEMP has also been the site for the large-scale demonstration of many new, innovative technologies ranging from vacuum removal insulation to "pipe explorers" to mobile work platforms. Some of these technologies have been used at other sites such as Oak Ridge, Hanford, Argonne and in Russia.

Ninety of the 273 site structures have been demolished through 2000. The most recent accomplishments have been the following:

- The maintenance/Tank Farm demolition was completed.

- Waste retrieval was accelerated from the Silos 1 and 2. The domes of their silos were sealed in order to reduce the release of radon. A contract was awarded for the accelerated remediation of Silo 3. (The silos are part of OU4.)
- Decontamination and dismantling has continued on Plant 5 (Metal Production Facility). This has included asbestos removal as well as the dismantling of equipment and systems.
- Decontamination and dismantling has been started on Plant 6 (Metal Fabrication Facility).

A unique feature of the FEMP is the creation of an On-Site Disposal Facility (OSDF), with a capacity of 1.9 million m^3. About 460 000 m^3 have already been filled. One main reason for the location of this permanent disposal facility on site is that 85% of the waste arising is estimated to be soil; only 15% will be decommissioning waste. However, many of the nuclides in this soil (classified as LLW) are long-lived ones so the future use of this site, which is not yet decided, will have to be with continuing federal government ownership.

It is interesting to note that the buildings are released on 250 µSv/year individual dose criterion to the public and that some of the buildings that are free released have higher activity contents than the cells of the disposal facility.

The other wastes arising are sent to the Nevada Test Site or to Envirocare in Utah.

Appendix 2

WORK OF THE TASK GROUPS

A2.1 Task Group on Decommissioning Costs

In 1989, the Co-operative Programme set up a Task Group on Decommissioning Costs in order to identify reasons for the large variations in reported cost estimates on decommissioning projects. The Task Group gathered cost data from 12 projects in the Co-operative Programme, established a basis for comparison of decommissioning tasks adopted in all projects, prepared a matrix of cost groups and cost items with a cost breakdown in "labour costs", "capital equipment and material" and "expenses", and incorporated the project cost data into a matrix.

One of the lessons learnt by the Task Group was the potential for making errors, and the difficulties encountered in performing quick international cost comparisons. It was evident that the answers to any cost questionnaire must be analysed and refined by follow-up questionnaires to understand the real contents. Another important observation the Task Group made was that there was no standardised listing of cost items or estimating methodology established for decommissioning projects. Such a standardisation would be useful not only for making cost comparisons more straightforward and meaningful, but should also provide a good tool for cost-effective project management. In their report, the Task Group made a proposal for a listing of cost items and cost groups that could be the framework for such a standardisation.

In November 1994, the Liaison Committee asked for the Task Group to be re-activated with the same objectives, looking this time (specifically and separately) at power reactors and fuel facilities. Quite early in the work of the re-started Task Group on Decommissioning Costs, it was noted that:

- The International Atomic Energy Agency (IAEA) was developing a technical document on cost of radioactive waste management and decommissioning of nuclear facilities, and had called international experts to form a Consultants Group on Decommissioning and Waste Management Costs.
- In its 1994-1998 Nuclear Fission Safety Programme, the European Commission (EC) decided to continue activities in view of setting up a database for decommissioning costs.

Based on the activities mentioned in the foregoing sections, a co-ordinated action was started with the three organisations (EC, IAEA, and OECD/NEA) in order to develop a common list of cost items for decommissioning operations recognising that such a standardised list would facilitate communication, promote uniformity and avoid inconsistency or contradiction of results or conclusions of cost evaluations for decommissioning projects carried out for specific purposes by different groups. As the work carried out by the Task Group on Decommissioning Costs of the OECD/NEA Co-operative Programme had advanced very well in the area, the work of the Task Group was used as the basis for further discussions.

A final report [1] was prepared describing the history, the scope, and the implementation of the co-ordinated action to develop a standardised list of decommissioning cost items and cost groups, including their respective definitions. It was published by the OECD/NEA in the first half of 1999 as a joint report from the three participating organisations (NEA, IAEA and EC).

Although it is hoped that the standardised list will be widely accepted and used, it was recognised that at that stage the list had achieved approval in theory only and should be further evaluated in practice. It was therefore proposed that the list be viewed as an interim version, to be broadly distributed, discussed and used, and to be finalised, most effectively in a workshop format, after approximately three years. At that point, a more definitive and more broadly tested and supported list should be issued as a report.

In the beginning of October 1999, after the publication of the common interim technical document, the Technical Advisory Group discussed the status of the work in the Task Group on Decommissioning Costs. It was recommended that a questionnaire should be send to the CPD projects to gather relevant data, however, to start analysing the collected data only if at least 5 questionnaires per group to be evaluated (nuclear power plants, fuel facilities) would respond to the enquiry. A final decision had been taken by the Liaison Committee during its meeting at the end of October 1999 that further work should be continued and that their participating projects intended to provide the data required in the questionnaire.

In spite of the extensive efforts of the Task Group, which produced a manual and supporting documentation to facilitate a response to the questionnaire and which also gave a presentation to the at the TAG-meeting in June 2000 in Knoxville to demonstrate the questionnaire, it had to be recognized at the end of April 2002 that sufficient data for a meaningful evaluation could not be obtained. As a result, the activities of the Task Group on Decommissioning Costs were terminated.

While the further work had been abandoned, the cost structure proposed in the interim report became the basis for an activity of the IAEA to prepare a technical document on “Decommissioning costs of WWER-440 nuclear power plants” [2]. The main objectives of this publication were to present the decommissioning costs of WWER-440 NPPs in a uniform manner, i.e. using the cost item and cost group system of the Interim Technical Document on Nuclear Decommissioning: *A Proposed Standardised List of Items for Costing Purposes* developed jointly by the EC, the IAEA and the OECD Nuclear Energy Agency (NEA), and providing, as such, a basis for understanding decommissioning costs differences.

Participants of the Task Group on Decommissioning Costs:

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A2.2 Task Group on Recycling and Re-use

Quite early during the information exchange, it became obvious that the management of the large volumes of contaminated materials arising from the decommissioning of nuclear facilities represents one of the most substantial cost fractions of such projects. Consequently, the minimisation of the volumes that must be disposed of as radioactive waste is a high priority goal for those undertaking decommissioning. It was also noted that much of this redundant material was valuable, e.g. stainless and other high quality steel or copper from wires. The recycling of such material (or its re-use or disposal) without radiological restrictions could be a significant means of achieving the aim of waste minimisation.

Therefore, in 1992, the Co-operative Programme set up a task group to study the recycling and re-use of redundant material from the decommissioning of nuclear facilities, in particular to provide information and insights into the practicality and usefulness of the criteria being developed for the release (clearance) of such material from regulatory control, seen from the perspective of organisations currently engaged in actual decommissioning operations.

The Task Group on Recycling and Re-use made a survey of the current practices and national regulations in this area, studied the technologies associated with recycling and analysed the proposed international recommendations and proposals for release criteria. The Task Group proposed a new approach to the release of materias from regulatory control, which differed from the guidance developed at IAEA and the EC. The group assessed the total health risks and compared the radiological risks associated with the recycling of material with the risk of disposing the material instead as radioactive waste and replacing it with new material. The results of this comparison showed that:

- radiological risks associated with both alternatives often can be very small in comparison with non-radiological industrial safety risks, and that
- non-radiological risks might be much lower for recycling because product manufacture starts from scrap metal and the risks associated with mining and refining of metal are avoided.

A report of the work of the Task Group was published in 1996 [3].

Following the release of this report, the Task Group, and in particular the Technical Secretary, focused on presenting its work as a contribution to the international debate in this area. In addition, the Task Group undertook to collect and analyse literature and material on this important subject, including a comparison of approaches used for the recycling and use of materials from the decommissioning of nuclear power plants and those used for materials from non-nuclear industries, also referred to as technologically enhanced naturally occurring radioactive material (TENORM).

The Task Group prepared a survey of the international discussions regarding clearance and exemption of materials from nuclear installation decommissioning operations and on the regulatory treatment of TNORM, and presented its work to the CPD in 2002 [4]. The survey, which includes numerous conference presentations, draft, interim and guidance materials for comment from regulatory authorities and international organisations, and statements from national standards or scientific academies related to these questions, showed that policy and regulatory approaches to the recycling and reuse of slightly contaminated materials were far from harmonised.

The Task Group also noted that management of radiological risk from materials from nuclear decommissioning and from TENORM follow different frameworks, which seem to apply higher burden to the nuclear decommissioners regarding the proof of compliance with safety goals.

Participants of the Task Group on Recycling and Reuse:

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A2.3 Task Group on Decontamination

Decontamination is a major decommissioning activity that may be used to accomplish several goals, such as reducing occupational exposures, permitting the reuse of components and facilitating waste management. The decision to decontaminate should be weighed against the total dose and cost.

In October 1992, the Technical Advisory Group of the Co-operative Programme established a Task Group on Decontamination in order to prepare a state-of-the-art report on decontamination in connection with decommissioning. The objective of this overview of decontamination techniques was to describe critical elements to be considered when selecting techniques for practical decontamination problems.

The work of the Task Group was focused on decontamination for dose reduction as well as for waste decategorisation. The decontamination of both metallic and concrete surfaces was considered.

During its early meetings, the Task Group developed a questionnaire which was sent to different project managers. The information requested in this questionnaire covered the technical as well as the economic aspects of selected decontamination techniques. The questionnaire was completed for each specific application of a given process, including actual data on efficiency of the process and on operating and investments costs.

Based on this questionnaire a list of decontamination processes was identified that may be used in connection with decommissioning. These processes have been divided into chemical, electrochemical and physical processes. Moreover, a distinction has been made between processes used in closed systems, e.g. full system decontamination of primary circuits or partial decontamination in a closed loop, and processes used in open systems, e.g. decontamination of dismantled pieces.

Only a limited number of questionnaires and relevant information was available. However it is felt that the information presented in the Task Group's final report represents a state-of-the-art view of the decontamination techniques available. The report confirms the importance of cost/benefit analyses very early in the process of selecting decontamination technologies for decommissioning to see if it is actually worth decontaminating the component or facility, or to determine whether a mild decontamination at low cost is more advantageous than an aggressive decontamination at a higher cost.

The draft was revised by the Task Group members, the TAG and the LC before a final version was published by the OECD/NEA [5].

The report gives an extensive and detailed overview of the data acquired for the survey. It presents a comprehensive list of real case examples for various decontamination techniques and processes applied in decommissioning.

Based on the information gathered some specific characteristics of selected decontamination techniques for segmented components and for building surfaces are discussed. In addition, some critical elements of choosing techniques for practical decontamination problems are given.

The information presented is not exhaustive. Practical experience in decontamination has also shown that a universal process does not exist. As such, future users should familiarise themselves with the characteristics of proposed techniques in order to make adequate choices based on specific requirements.

While the report itself is freely available, the appendix with the data gathered has been treated on a confidential basis, and was only distributed to members of the Task Group, the TAG and the LC.

Participants of the Task Group on Decontamination:

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A2.4 Task Group on Release Measurements

The Task Group on Release Measurements was established in December 1996, after a recommendation by the Task Group on Recycling and Re-use, that a specialist group should study the problems that arise in connection with activity measurements at the extremely low levels.

The strict application of these allowable dose levels, without any broader consideration of other non-radiological risks that could be avoided by recycling, has led to recommendations of extremely low permitted activity levels in material to be released from regulatory control.

For applying these recommendations effectively, adequate methods of measurement must be available to demonstrate or verify that the activity levels are lower than the proposed levels. Measurements would have to be made under practical industrial conditions, where various constraints could significantly influence the results. The costs of activity measurements at extremely low levels on large quantities of equipment with complex geometries could be prohibitively high.

The Co-operative Programme therefore established the Task Group on Release Measurement to study these problems in an analytical and structured manner. The terms of reference for the Task Group were briefly:

- Make an overview of the available measurement techniques at release levels.
- Study the limitations and constraints of using these techniques on an industrial scale.
- Consider financial aspects for implementing of measurement methods.

A first draft included technical chapters and a critical discussion on methods and techniques, but did not have the originally planned chapter on "costs of release measurements". This was because of the low response from projects (and other sources) to questions in this area due to the fact that this information is considered to be confidential.

The LC requested the Task Group to make a special effort to complete the report as originally planned (with cost data) because of its importance in clearance and (generally) decommissioning discussions. The Group approached specific projects and collected relevant data, which was used to write the cost chapter. The final report [6] of the Task Group has been released for publishing.

Participants of the Task Group on Release Measurements:

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