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NUCLEAR ENERGY AGENCY

Radioactive Waste Management Committee (RWMC)

Management Board of the Co-operative Programme for the Exchange of Scientific and Technical Information concerning Nuclear Installations Decommissioning Projects

THE NEA CO-OPERATIVE PROGRAMME ON DECOMMISSIONING

PROGRESS DURING 1995-2005

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FOREWORD

The NEA Co-operative Programme for the Exchange of Scientific and Technical Information Concerning Nuclear Installation Decommissioning Projects (CPD) is a joint undertaking of a limited number of organizations, mainly from NEA member countries. The objective of the CPD is to acquire information from operational experience in decommissioning nuclear installations that is useful for future projects. The working method is through sharing of scientific and technical knowledge drawn from a number of current decommissioning projects. Initiated in 1985, the CPD is administratively organized based on its formal, signed Agreement that was renewed in 1990 and 1995. The fourth five year period commenced in 2000 with the Agreement then subject to major revision and was not formally agreed and signed until April 2004.

This report describes the progress and the generic results obtained by the CPD during 1995-2005. Although part of the information exchanged within CPD is confidential in nature and is restricted to programme participants, experience of general interest gained under the programme is released for broader use. Such information is brought to the attention of all NEA members through regular reports to the Radioactive Waste Management Committee (RWMC) of the NEA, as well as through experience summary documents such as this report. The Working Party on Decommissioning and Dismantling (WPDD) within RWMC would like to thank CPD for sharing the experiences from its important work.

Similar reports have been published at the end of the first and second five year periods as NEA reports. This current report is another in the same series of reports and is thus a successor to “The NEA Co-operative Programme on Decommissioning: The First Ten Years 1985 – 1995”.

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BACKGROUND

The Co-operative Programme on Decommissioning promoted by the Nuclear Energy Agency (NEA) of the OECD has recently completed 20 years of operation. The Programme, which is essentially a scientific and technical information exchange programme between decommissioning projects, came into being in 1985, under a five year agreement between the participating organizations. Its demonstrated usefulness led to the renewal of the agreement to new five year agreements first in 1990 then in 1995. The current agreement runs from 2004 to 2009. The Co-operative Programme initially consisted of 10 decommissioning projects in seven countries. The programme has since grown to 41 projects in 11 NEA member countries and one non-member economy (December 2005).

The Co-operative Programme on Decommissioning (CPD) is a joint undertaking and functions within the framework of an agreement between 21 organisations actively executing, planning or having plans regarding decommissioning of nuclear facilities. The objective of 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 available experience is available and that safe, environmentally friendly and cost effective methods are employed in all decommissioning projects.

The Co-operative Programme on Decommissioning (CPD) informs and reports to the NEA Radioactive Waste Management Committee. The Programme itself is governed by a Management Board, where all participants are represented. The main forum for the exchange of information is the Technical Advisory Group. Task Groups have been set up to study specific topics of common interest as needed and proved being an efficient instrument to achieve the goals of the CPD agreement. A key factor in the success and utility of this programme has been the unanimous agreement of participants that information and experience exchange regarding installations and projects are based on a “give and take” approach.

Projects Participating in the Co-operative 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 25 reactors, 8 reprocessing plants and 8 fuel facility projects Some general comments are given below.

- 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 HTGRs both with block type and pebble bed fuel design. The list of reactor projects includes also the decommissioning of a plant with two former Russian training reactors.
- Of the 25 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 the sites have been returned to “green field conditions” or have been decontaminated completely so as to have been removed from regulatory control. The completed projects continue to be considered as being part of the Programme, and CPD still benefit from the experience of these projects. The dormant plants can continue to generate information and experiences on building/plant degradation and long-term surveillance.

- 27 of the 40 plants in the Programme are to be, or have been decommissioned to Stage 3, namely total dismantling and decontamination.
- Many of the earlier projects in the Programme were concerned with experimental or prototype plants. The projects which have joined the Programme at a later date have, for understandable reasons, concerned 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.

Some Lessons Learned during CPD 20 years of work

- 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 experience on design, construction and operation is a considerable help for reliable planning, cost evaluation and successful realization 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 organizations.

Technical Advisory Group

The central instrument 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 projects. As the Programme Agreement contains provisions and conditions protecting the information exchanged by restricting its use and release, discussions at the TAG are free and open. Because of this frank and open nature of the discussions, questions and answers at the TAG meetings, measures has been taken to ensure that information they consider as restricted continues to be protected.

During the years that have passed since the start of the Co-operative Programme, there has been a large increase in the number of projects. At the same time the representatives of the projects have changed periodically.

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, among others, was gained, on alpha penetration of concrete and characterisation. This was seen as being very beneficial to JRTF Japanese decommissioning projects. During one year, in the 1990's a Japanese colleague stayed, as a trainee, in 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 development, estimating and planning of decommissioning operations. In addition information relating to waste management and size 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 approach to the decommissioning of 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 resulted in this approach being considered by the Canadian government who have since created a LMU.

Feedback on generic Experience - 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 information 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 incorporated the gathered information into this 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. Numbers taken at face value, without regard to their context, are easily misunderstood and misinterpreted.

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 (LC) 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 the 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 with the objective setting up a database for decommissioning costs.

Based on these concurrent activities and their similar aims and on the initiative of the Co-operative Programme, the CPD, IAEA and EC agreed to start a co-ordinated action in order to produce a standardised or uniform listing of cost items and related cost-item definitions for decommissioning projects. Such a standardised list as described previously, 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.

Work on the co-ordinated action has resulted in a new uniform and complete approach to decommissioning costs, which has been presented in an interim technical report, “A Proposed Standardised List of Items for Costing Purposes”, jointly published by the NEA, the IAEA and the EC. Input to the report has been from experts representing the three organisations and their supporting groups. It is recognised that at this stage the listing in the report is accepted and is issued for use. Following from the use of this standardised costing list for some years it is suggested that the NEA, IAEA & EC revisit the topic and undertake any revisions following consideration of the experience to date. This has not happened yet as there has not been sufficient experience to justify reconsideration (lack of data).

The Task Group prepared a questionnaire based on the standardised cost item list as well as a manual to help projects to complete the questionnaire. Returned questionnaires were used to update the standardised costing list.

Feedback on generic Experience - Recycling and Re-use

Quite early during the information exchange, it became obvious that the management of the large volumes of contaminated (slightly) 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 have to be disposed 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. The recycling of such material (or its reuse or disposal) without radiological restrictions could be a significant means of achieving the aim of waste minimisation. So, as a result of the information exchange, in 1992 the Programme set up a task group to study the recycling and reuse of redundant material from the decommissioning of nuclear facilities.

The Co-operative Programme's Task Group on Recycling and Reuse made a survey of the current practices and national regulations in this area, studied the technologies associated with recycling and analysed internationally gathered data for release criteria. A report of the work of the Task Group was published in 1996.

In the report in addition to characterising the radiological risks associated with the release of material, the Task Group assessed the total health risks, comparing 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 show that

- The report points at the problem that radiological risks associated with both alternatives often can be very small in comparison with non-radiological industrial safety risks,
- The non-radiological risks might be much lower for recycling because product manufacture starts from scrap metal. The risks associated with mining and refining of metal are in this case avoided. These issues have been discussed in a study on recycling and re-use performed between SSI, USDOE (ANL), CEA (IPSN), STUDSVIK, Belgoprocess and Åkers AB with the title "Validation of dose calculation programmes RESRAD (USA)/CERISE (France)"

Feedback on generic Experience - 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. The work of the Task Group was focused on decontamination for dose reduction as well as for waste de-categorisation. The decontamination of both metallic and concrete surfaces was considered. The objective of this overview of decontamination techniques was to describe critical elements to be considered when selecting techniques for practical decontamination problems.

Based on a questionnaire which was sent to the various project managers, a list of decontamination processes has been 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.

A limited number of questionnaires and relevant information could be made available. Based on this information, a draft final report of the Task Group was prepared, commented by the Task Group members, the TAG and the LC before a final version was published by the OECD/NEA.

Feedback on generic Experience - Release Measurements

The Task Group on Release Measurements was established in December 1996, after a recommendation by the Task Group on Recycling and Reuse, 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 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 final draft of the Task Group's report, except for the conclusions, was presented to the LC in October 1999. The 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 Task Group report has been submitted to the NEA for final check and publication is to be produced.

Significance of the Programme for the Participants

As mentioned earlier in the document, 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 nuclear 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 from country to country, indeed even in different parts of the same country.

So the knowledge and information gleaned from the Programme is both generally applicable and of common interest at one level and is project specific at another. Even in the case of project specific problems, the administrative approaches and manner in which they are solved.The main forum of information exchange between the projects has been, as stated earlier, the TAG meetings. Another important functional area of the Co-operative Programme has been the work in the various task groups, particularly the Task Group on Recycling and Reuse and the Task Group on Decommissioning Costs. The results of the work of the task groups have, at a drafting stage, been discussed and analysed at meetings of the TAG. Apart from the technical information exchange that has been a great value to participants; the following general aspects can be underlined.

Decommissioning of nuclear installations will become, in a few decades one of the important sectors in the nuclear market probably being of similar importance to the design, construction and operation of nuclear installations. In any case the lessons learned during decommissioning have to be made available in the nuclear field in order to improve design, construction and operation of future nuclear installations.

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. This will help those undertaking decommissioning to make reliable plans and sound cost evaluation and will improve safety. This latter aspect is essential for countries with limited resources for decommissioning their nuclear installations.

In addition to the tangible benefits listed above is the 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. This is an invaluable asset.

Future of the Co-operative Programme

The OECD Nuclear Energy Agency’s Co-operative Programme on Decommissioning was initiated on the basis of a proposal from the USDOE in 1985. Its main purpose was and still is the exchange of technical and scientific information arising from the planning and execution of major decommissioning projects on nuclear facilities. Starting with the modest number of 10 projects from 7 countries, it has grown to be the major forum in the world for this purpose. This is clearly reflected by the fact that today the Programme has 41 projects from 12 participating members as participants and several new projects are knocking on the door for admission.

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 its current form, based on confidentiality, and a “give and take” approach. The organisation of the TAG meetings may have to be modified to adjust to the larger number of participating projects, but, as can be observed at present, the current procedures are not yet at a point of saturation.

Looking back over the twenty years since the OECD Nuclear Energy Agency established the Co-operative Programme on Decommissioning the Programme has functioned as the main international forum for the exchange of technical, scientific and other information arising from nuclear decommissioning projects. During these years, nuclear decommissioning has grown from local specialist activities to a competitive commercial industry.

In specific areas of common interest, the results of the various task groups of the Co-operative Programme are being used to further the interests of nuclear decommissioning.

1. INTRODUCTION

The Co-operative Programme on Decommissioning under the Nuclear Energy Agency (NEA) of the OECD has recently completed 20 years of operation. The Programme, which is essentially an information exchange programme between decommissioning projects, came into being in 1985, under a five year agreement between the participating organizations. Its demonstrated usefulness led to the renewal of the agreement to new five year agreements first in 1990 then in 1995. During this period it has grown from an initial 7 participating members with 10 decommissioning projects to 14 participating members with 39 decommissioning projects.

Whilst it was decided to continue with the Co-operative Programme following the completion of the five year programme in 2000 the terms of the agreement were renegotiated and were duly agreed and signed in April 2004. Under this revised agreement initially there were 12 participating members with 40 decommissioning projects.

Formally titled the “Co-operative Programme for the Exchange of Scientific and Technical Information Concerning Nuclear Installation Decommissioning”, the roots of the Programme go 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 Radioactive Waste Management Committee of the NEA 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 CPD is a programme for decommissioning projects from member organisations of the OECD. However, in specific cases, and with notification to the NEA Steering Committee, 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 available 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 succeossor to the earlier Liaison Committee, where all participants are represented. The main forum for the exchange of information is the Technical Advisory Group, which meets twice a year. Groups are set up to study specific topics of common interest. 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.

The Programme has published two overview five-year reports earlier [2,3]. This report will concentrate on the last ten years and describe the activities of the Management Board, the Technical Advisory Group and the various Task Groups during that period. It will describe how the Programme and its activities and procedures have evolved over the years and indicate 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 overview the achievements of the Co-operative Programme and visualise future developments in the field

- Compendium on Decommissioning Activities in NEA Member Countries. UNC Nuclear Industries (supported by the United States Department of Energy)
  OECD/NEA 1985
- International Co-operation on Decommissioning. Achievements of the NEA Co-operative Programme.
  OECD/NEA 1992
- The NEA Co-operative Programme on Decommissioning. The First Ten Years.
  OECD/NEA 1996.

2. STRUCTURE OF THE CO-OPERATIVE PROGRAMME

2.1 General

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. 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. In addition to these “owners” of such projects, 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 known as UNIPEDE, participate as observers, who give and receive general and overview information on programmes projects and task group activities.

2.1.1 The membership of the Co-operative Programme as of December 2003 were as follows:

Organisations Participating in the Programme

Participating Organisations	
• Belgoprocess n.v. • Centre d’étude de l’énergie nucléaire/Studiecentrum vor Kernergie (CEN•SCK)	Belgium
• Atomic Energy of Canada Limited/Énergie atomique du Canada limitée (AECL/AECL)	Canada
• Alara A.S.	Estonia
• Kernkraftwerk Lingen GmbH • Forschungszentrum Karlsruhe (FzK) GmbH • Energiewerke Nord GmbH (EWN) • Wiederaufarbeitungsanlage Karlsruhe (WAK) GmbH • Arbeitsgemeinschaft Versuchsreaktor (AVR) GmbH	Germany
• Commissariat à l’énergie atomique (CEA) • Électricité de France (EDF) • AREVA/COGEMA	France
• SOGIN p.a. • Comitato Nazionale per la Ricerca e per lo Sviluoppo dell’Energia Nucleare e delle Energi Alternative (ENEA)	Italy
• Japan Atomic Energy Research Institute (JAERI) • Japan Nuclear Cycle Development Institute (JNC)	Japan
• Korea Atomic Energy Research Institute (KAERI)	Republic of Korea
• SE-VYZ Bohunice	Slovak Republic
• Centro de Investigaciones Energeticas Medioambientales y Tecnolgicas (CIEMAT) • Empresa Nacional de Residuos Radioactivos SA (ENRESA)	Spain
• Svensk Kärnbränslehantering AB (SKB)	Sweden

• Institute of Nuclear Energy Research	Chinese Taipei
• United Kingdom Atomic Energy Authority (UKAEA) • British Nuclear Fuels PLC (BNFL)	United Kingdom
• Department of Energy (USDOE) • Public Services of Colorado	United States

Observers are also invited to attend meetings.

Invited Observer Organisations	
• European Commission (EC) • International Atomic Energy Agency (IAEA) • EURELECTRIC (formerly UNIPEDE)	

Note: The organisations listed above are those with projects in the Programme or those associated as observers. In addition, there are a number of organisations who contribute to the Programme by assigning specialists to the various task groups and special arrangements.

2.1.2 The continued membership of the Co-operative Programme is as follows:

Organisations Continuing to Participate in the Co-operative Programme Under the March 2004 Agreement

• Centre d'etude de l'energie nuclaire/Studiecentrum vor Kernergie (CEN•SCK) • Belgoprocess	Belgium
• Atomic Energy of Canada Limited/ Énergie atomique du Canada limitée (AECL/AECL)	Canada
• Commissariat à l'énergie atomique (CEA) • AREVA/COGEMA • CODEM • Électricité de France	France
• Forschungszentrum Karlsruhe GMBH (FZK) • Energiewerke Nord GmbH (EWN) • Wiederaufarbeitungsanlage Karlsruhe GmbH (WAK) • Arbeitsgemeinschaft Versuchsreaktor GmbH (AVR)	Germany
• SOGIN p.a.	Italy
• Japan Atomic Energy Research Institute (JAERI) • Japan Nuclear Cycle Development Institute (JNC) • JAPCO • RANDEC	Japan
• Korea Atomic Energy Research Institute (KAERI)	Republic of Korea
• SE-VYZ Bohunice	SlovakRepublic
• Centro de Investigaciones Energeticas Medioambienty Tecnolgicas (CIEMAT) • Empresa Nacional de Residuos Radioactivos SA (ENRESA)	Spain
• Svensk Kärnbränslehantering AB (SKB)	Sweden
• Institute of Nuclear Energy Research	Chinese Taipei
• United Kingdom Atomic Energy Agency (UKAEA) • British Nuclear Fuels PLC (BNFL)	United Kingdom

The Co-operative Programme is implemented by two groups: a governing body and a technical group. The governing body, called the Management Board (CPDMB), comprises representatives of all participants, including the observers. It is responsible for the general conduct and orientation of the programme, including direction and supervision of the work programme, establishment of criteria for dissemination of the information exchange or generated within the Programme, approval of changes of membership, etc. The secretariat of the CPDMB is ensured directly by the NEA Secretariat. The CPDMB meets once a year. A CPDMB 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.

The 40 projects in the Programme as of December 2004 include the decommissioning of 24 reactors, 8 reprocessing plants, 7 fuel material plants, and 1 isotope handling facility. A full list of those projects currently participating in the Co-operative Programme is provided in tabular form at the end of this paper.

2.2 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 member projects. 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, 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.

During the years that have passed since the start of the Co-operative Programme, there has been a large increase in the number of projects. At the same time the representatives of the projects have changed periodically. As a result of these developments, there was a general expression of need among the participants of the TAG 23 meeting (Chester, United Kingdom in October 1997) for information on the Programme, its original objectives, its evolution, its future, etc. On the initiative of the TAG Chairman, the current understanding has been laid out in a document describing the scope and the objectives of the Programme, the relationship that the Programme and its organisation has with the OECD/NEA, as well as detailing the roles that the various bodies, groups and officials play in achieving these the Programme’s objectives.

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 has to 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.

2.3 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 making such studies/analyses. Task groups have worked in the following areas:

- ***Decommissioning costs.*** A first study was made in 1989-1991. A renewed survey and study was requested by the LC in 1994. This new Group was created accordingly in 1995, and has now completed its work. The Group has noted that it nuclear decommissioning is increasingly a commercial industry. One consequence of this is that there is greater reluctance to publish commercially interesting information such as costs or details of specific methods such as for decontamination.

  The Task Group on Decommissioning Costs has also noted that similar exercises to its own are ongoing in several other areas, such as the chemical and defence related industries, and that other agencies, such as the International Atomic Energy Agency and the European Commission, are also working in this area

- ***Recycling and reuse*** of slightly contaminated redundant material from decommissioning. This Task Group has been ongoing for some time, and has observed that huge volumes of very low-level radioactive material arise in a large number of non-nuclear industries. The Group has highlighted its views, within the NEA, on the issue of consistency in the regulatory treatment of such radioactive materials, irrespective of their industrial origin.

- ***Decontamination*** in connection with decommissioning. This Task Group was established in order to prepare a state-of-the-art report on decontamination in connection with decommissioning, including a description of critical elements to be considered when selecting techniques for a practical decontamination procedure. Information requested covered the technical aspects as well as the economic aspects of selected decontamination techniques. A limited number of questionnaires and relevant information could be made available.

- ***Release measurements.*** This Task Group was established in order to study the problems that arise in connection with activity measurements at the extremely low levels required by the existing draft/interim release criteria. A first draft report with technical chapters and a critical discussion on methods and techniques is currently revised in order to add a chapter on "costs of release measurements".

The Co-operative Programme is basically a volunteer activity. Participation in a Task Group often involves 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.

2.4 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 the US RESRAD and the French CERISE Codes. The project was initiated and is managed by the Swedish Radiation Protection Institute (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

2.5 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 TAG and the LC (the predecessor of the CPDMB) 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 in the latest five year agreement for the continuation of the Co-operative Programme a financial contribution from each member is sought.

2.6 Link to the NEA Committee Structure

As mentioned earlier, 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 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 Radioactive Waste Management Committee. The Programme itself is governed by a Management Board, where all participants are represented. The main forum for the exchange of information is the Technical Advisory Group, which meets twice a year. The members of CPD 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 there is an exchange of members both on the relevant committees and Bureaux as well as an interface document.

There has been increasing awareness, partly through the information from the Co-operative Programme, that decommissioning and dismantling technology have become mature. Overall issues of decommissioning policy including waste management, and the implications of decommissioning on the sustainability of nuclear power need to be highlighted. It has also been recognised that there are common aspects of material management and implementation in the area of both waste management and decommissioning, which benefit from open implementer/regulator dialogue.

References

(1) Co-operative Programme for the Exchange of Scientific and Technical Information Concerning Nuclear Installation Decommissioning Projects. Mission Statement, NEA/CPD/DOC(98)4 1998.

3. PROGRAMME ACTIVITIES

3.1 Projects Participating in the Co-operative Programme

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 participating projects is given in the tables on the following pages. For these projects some general comments are given below:

- 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 HTGRs both with block type and pebble bed fuel design. The list of reactor projects also includes the decommissioning of a plant with two Russian submarine reactors.
- Of the 24 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 reckoned 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 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 facility.
- 27 of the 40 plants in the Programme are to be, or have been 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 have, for understandable reasons, concerned 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 each project are briefly described in Annex I.

Table 3.1 Completed Reactor Projects at December 2003

Facility	Type	Operation	Decommissioning	Power or throughput	Project time-scale	Cost-estimate	Entry into Programme	Remarks
1. Gentilly 1 Canada	Heavy-water moderated/ boiling light-water-cooled prototype	1972-78	Variant of Stage 1	250 MWe	1984-1986	MCAD 25 (1986)	1985	In dormancy
2. NPD Canada	PHWR CANDU prototype	1962-87	Variant of Stage 1	25 MWe	1987-1988	MCAD 25.3	1988	In dormancy
3. Rapsodie Cadarache France	Experimental sodium-cooled fast-breeder reactor	1967-82	Stage 2	20 MWt	1983-1994	MFRF 131.7 (1989)	1985	In dormancy
4. G2/G3 Marcoule France	GCR, Electricity and nuclear materials production	1958-80	Stage 2	250 MWt each	1982-1993	MFRF 150 (1990)	1985	Stage 2 achieved
5. KKN Niederaichbach Germany	Gas-cooled/ heavy-water moderated	1972-74	Stage 3	106 MWe	1988-1994	MDEM 190 Total costs: MDEM 269	1985	Fixed – price contract Stage 3 achieved
6. KWL Lingen Germany	BWR (with superheater)	1968-77	Stage 1	520 MWt	1985-1988	-	1985	In dormancy
7. HDR Germany	BWR, nuclear superheat	1969-71	Stage 3	100 MWt	1993-1998	MDEM 100	1993	Stage 3 achieved
8. JPDR Tokai Japan	BWR	1963-76	Stage 3	90 MWt	1986-1996	MJPY 22 500	1985	1981-1986 R&D Stage 3 achieved
9. Shippingport United States	PWR	1957-82	Stage 3	72 MWe	1985-1989	MUSD 91.3 (1990)	1985	Fixed – price contract Stage 3 achieved **Project no longer in the CPD Programme**
10. EBWR United States	BWR	1956-67	Stage 3	100 MWt	1986-1996	MUSD 19.4	1990	**Project no longer in the CPD Programme**
1. Fort St Vrain United States	HTGR	1976-89	Stage 3	330 MWe	1972-1995	MUSD 174	1993	Fixed – price contract **Project no longer in the CPD Programme**

Table 3.2 **Reactor Projects in Progress at December 2003**

	Type	Operation	Decommissioning	Power or throughput	Project time-scale	Cost-estimate	Entry into Programme	Remarks
1. BR-3, Mol Belgium	PWR	1962-87	Stage 3 (Partial)	41MWt	1989-2010	150MEuro (2000)	1988	EC Pilot Project
2. Paldiski Estonia	Soviet submarine PWR	1966-85	Stage 1		1994-		1997	**Project no longer in the programme**
3. EL4 France	Gas cooled/heavy-water moderated	1966-85	Stage 2	70 MWe	1989-1999	MFRF 550 (1995)	1993	-
4. MZFR, Karlsruhe Germany	PWR Heavy-water-cooled and moderated	1965-84	Stage 3	50 MWe	1984-2005	MDEM 440	1989	-
5. Greifswald Decommissioning Project, Germany	VVER	1973-90	Stage 3	8x 440 MWe	-	-	1992	-
6. AVR Germany	Pebble bed HTGR	1967-88	Stage 3	15 MWe	-	-	1994	Stage 3 being planned
7. KNK, Karlsruhe Germany	Fast breeder reactor	1971-91	Stage 3	20 MWe	1991-2003	500 MDEM	1997	-
8. Garigliano Italy	BWR (Dual cycle)	1964-78	Stage 3 planned by 2020	160 MWe		297 MEuro (2000)	1985	In 1999 decommissioning strategy changed from SAFSTOR to DECON
9. Latina Italy	GCR (Magnox)	1963-86	Stage 3 planned by 2020	210/160 MWe		615 MEuro (2000)	1999	“
10. Fugen Japan	Light water cooled Heavy water moderated	1979-2003	Stage 3	165 MWe	2003-		2000	Advanced Thermal Reactor
11. KRR-1 & 2 Korea	Pool type research reactors (Triga 1 & 2)	1962-95 1972-95	Stage 3	250 KWt 2 MWt	1997-2008		1997	
12. Bohunice A1 Project Slovak Rep.	Gas cooled, heavy-water-moderated	1972-79	Stage 1	150 MWe			1992	Decommissioning after fuel accident
13. Vandellos 1 Spain	GCR	1972-89	Stage 2	500 MWe	1992-2000	MESP 14,600	1993	
14. Taiwan Research Reactor Chinese Taipei	Light water cooled Heavy water moderated	1973-88	Partial dismantling	40 MWt	1998-2002			Remodelling as multipurpose reactor
15. WAGR, Sellafield UK	AGR	1962-81	Stage 3	100 MWt	1983-1998	MGBP 58	1985	EC Pilot project
16. Prototype Fast Reactor PFR, Dounreay, UK	Sodium cooled fast breeder reactor	1974-94	Stage 1	250 MWe			1997	

Table 3.3 Completed Fuel Facility Projects at December 2003

Facility	Type	Operation	Decommissioning	Power or throughput	Project time-scale	Cost-estimate	Entry into Programme	Remarks
1. Tunney's Pasture Facility, Ottawa Canada	Isotope handling facility	1952-83	Stage 3	-	1990-94	MCAD 13 (1991)	1990	Stage 3 achieved
2. BNFL Co-precipitation Plant, Sellafield UK	Production of mixed plutonium and UO_2 fuel	1969-76	Stage 3	50 Kg/d	1986-90	KGBP 2,245 (1990)	1987	Stage 3 achieved

Table 3.4 **Fuel Facility Projects in Progress in December 2003**

Facility	Type	Operation	Decommissioning	Power or throughput	Project time-scale	Cost-estimate	Entry into Programme	Remarks
1. Eurochemic Reprocessing Plant, Dessel Belgium	Reprocessing of fuel	1966-74	Stage 3	300 Kg/d	1989-2012 (Main process building)	179 MEUR (2003)	1988	Execution by in-house staff
2. Building 204, Bays Project Chalk River, Canada	Fuel storage pond	1947 to date					1997	
3. AT-1, La Hague France	Pilot reprocessing plant for FBR	1969-79	Stage 3	2 Kg/d	1982-1998	MFRF	1985	EC Pilot Project
4. Radio Chemistry Laboratory, Fontenay-aux-Roses France	Reprocessing R&D	1961 -95	Stage 3		1995-2011		1999	
5. WAK Germany	Prototype reprocessing plant	1971-90	Stage 3				1993	
6. JRTF, Tokai Japan	Reprocessing test facility	1968-70	Stage 3		1991-2004	MJPY 8,600	1991	
7. ACL Project,Studsvik AB Sweden	PU & enriched fuel research	1963-97	Stage 3		1998		1999	
8. BNFL, 204 Primary Separation Plant Sellafield, UK	Reprocessing facility	1952-73	Stage 3	Metal = 500 t/a Oxide = 140 t/a	1990-2010	MGBP 90	1990	
9. West Valley Demonstration Project United States	Reprocessing plant for LWR fuel	1966-72	Stage 3	100 t/a	1982-2024	MUSD 1,400	1986	**This project is no longer in the programme.**
10. FEMP United States	Hexafluoride reduction plant	1954-56	Stage 3				1993	**This project is no longer in the programme**

Table 3. 5 **Reactor Projects in Progress in December 2004**

Facility	Type	Operation	Decommissioning	Power or throughput	Project Time-scale	Cost-estimate	Entry into Programme	Remarks
1. BR-3, Mol Belgium	PWR	1962-87	Stage 3 (Partial)	41MWt	1989-2010	150MEuro (2000)	1988	EC Pilot Project
2. Gentilly 1 Canada	Heavy water moderated boiling light water cooled	1967-82	Variant of Stage 1	250MWe	1984-86	MCAD 25 (1986)	1985	In dormancy
3. NPD Canada	PHWR CANDU	1967-87	Variant of Stage 1	25MWe	1987-88	MCAD 25.3	1988	In dormancy
4. Bugey 1 France	Gas graphite reactor	1972-94	Stage 3	540 MWe	1997-2021		2004	
5. G2/G3, Marcoule France	GCR	1958-80	Stage 2	250 MWe each	1982-1993	MFRF 150 (1990)	1985	Stage 2 achieved
6. Melusine France	Pond research reactor	1988-93	Stage 3	8MWt	1999-2006	20MEuro (2003)	2004	
7. Rapsodie, Cadarache France	Experimental sodium cooled fast breeder reactor	1967-82	Stage 2	20MWt	1983-1994	MFRF 131.7 (1989)	1985	In dormancy
8. KKN, Neideraichbach Germany	Gas cooled/heavy water moderated	1972-74	Stage 3	106 MWe	1988-1995	MDEM 135	1985	Stage 3 achieved
9. MZFR, Karlsruhe Germany	PFR Heavy water cooled and moderated	1965-84	Stage 3	50 MWE	1994-2005	MDEM 440	1989	
10. Greifswald Decommissioning Project, Germany	VVER	1973-90	Stage 3	8x440 MWe			1992	
11. AVR Germany	Pebble bed HTGR	1967-88	Stage 3	15 MWE			1994	
12. KNK, Karlsruhe Germany	Fast breeder reactor	1971-91	Stage 3	20 MWE	1991-2003	MDEM 500	1997	
13. HDR Germany	BWR, nuclear superheat	1969-71	Stage 3	100 MWt	1994-1998	MDEM 50	1993	Stage 3 achieved
14. Garigliano Italy	BWR (Dual cycle)	1964-78	Stage 3 planned by 2020	160 MWe		297 MEuro (2000)	1985	
15. Latina Italy	GCR (Magnox)	1963-86	Stage 3 planned by 2020	210/160 MWe		615 MEuro (2000)	1999	
16. JPDR, Tokai Japan	BWR	1963-76	Stage 3	90 MWe	1986-1996	MJPY 22,500	1985	1981-1986 R & D. Stage 3 achieved

Table 3. 5 (cont.) **Reactor Projects in Progress in December 2004**

Facility	Type	Operation	Decommissioning	Power or Throughput	Projects Time-scale	Cost-estimate	Entry into programme	Remarks
17. Fugen Japan	Light water cooled. Heavy water reactor	1979-2003	Stage 3	165 MWe	2003-2023		2000	
18. Tokai 1 Japan	GCR	1966-1998	Stage 3	166 MWe	2001-2017	660MEuro (2004)	2002	Timetable depends on availability of repository
19. KRR 1 & 2 Korea	Pool type research reactors	1962-95 1972-95						
20. Bohunice A1 Slovakia Rep.	Gas cooled, heavy water moderated	1972-79	Stage 1	150 MWe			1992	
21. Vandellos 1 Spain	GCR	1972-89	Stage 2	500 MWe	1992-2899	MESP 14,600		
22. Taiwan Research Reactor Chinese Taipei	Light water cooled Heavy water moderated	1973-88	Partial dismantling	40 MWt	1998-2002			
23. WAGR, Sellafield UK	AGR	1962-81	Stage 3	100 MWt	1983-1998	MGBP 58	1985	
24. Prototype Fast Reactor PFR, Dounreay UK	Sodium cooled fast breeder reactor	1974-94	Stage 1	250 MWe			1997	

Table 3.6 **Fuel Facility Projects in Progress in December 2004**

Facility	Type	Operation	Decommissioning	Power or throughput	Project time-scale	Cost-estimate	Entry into Programme	Remarks
1. Eurochemic Reprocessing Plant, Dessel Belgium	Reprocessing of fuel	1966-74	Stage 3	300 Kg/d	1989-2012 (Main process building)	179 MEUR (2003)	1988	Execution by in-house staff
2. Building 204, Bays Project Chalk River, Canada	Fuel storage pond	1947 to date					1997	
3. Tunney's Pasture Facility Ottawa, Canada	Isotope handling facility	1952-83	Stage 3		1990-1994	MCAD 13 (1991)	1990	Stage 3 achieved
4. AT-1, La Hague France	Pilot reprocessing plant for FBR	1969-79	Stage 3	2 Kg/d	1982-1998	MFRF	1985	EC Pilot Project
5. Radio Chemistry Laboratory, Fontenay-aux-Roses France	Reprocessing R & D	1961 -95	Stage 3		1995-2011		1999	
6. ATUE France	Recovery of enriched uranium	1965-96	Stage 3		2000-07	30 MEuro	2004	
7. Elan IIB France	Manufacture of ^{137}Cs & ^{90}Sr sources	1970-73						
8. APM, Marcoule France	Pilot reprocessing plant	-97	Stage 3		-2012	450MEuro	2004	
9. UP1, Marcoule France	Industrial reprocessing plant	1958-97	Stage 2	500 t U/a	1998-2028	2,000 MEuro	2002	Main project in France
10. WAK Germany	Prototype reprocessing plant	1971-90	Stage 3				1993	
11. JRTF, Tokai Japan	Reprocessing test facility	1968-70	Stage 3		1991-2004	MJPY 8,600	1991	
12. Plutonium Fuel Fabrication Facility Japan	Fabrication of MOX fuels	1972-2002 (for ATR) 1972-1988 (ex.FBR)	Stage 3	10t MOX /a 1t MOX/a	-2020 (excluding building)		2004	
13. Uranium Conversion Facility Korea	Conversion of yellowcake to UO_2/UF_4	1982-92	Stage 3	100 t U/a	2000-2007		2003	
14. ACL Project,Studsvik AB Sweden	PU & enriched fuel research	1963-97	Stage 3		1998		1999	

Table 3.6 (cont.) **Fuel Facility Projects in Progress in December 2004**

Facility	Type	Operation	Decommissioning	Power or throughput	Project time-scale	Cost-estimate	Entry into Programme	Remarks
15. BNFL 204 Primary Separation Plant Sellafield, UK	Reprocessing facility	1952-73	Stage 2	Metal = 500 t/a Oxide = 140 t/a	1990-2010	MGBP 90	1990	
16. BNFL Co-precipitation Plant, Sellafield UK	Production of mixed plutonium and UO_2 fuel	1969-76	Stage 3	50 kg/d	1986-1990	KGBP 2,245 (1990)	1987	Stage 3 achieved

3.2 Directions in the Development in Decommissioning Projects

As can be seen from the tables 3.1 – 3.6 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. Some of these are indicated in the following paragraphs:

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 owing 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.

Sequential licensing/granting of funding

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 a single license issued at the start of the project. Thus MZFR is being decommissioned under 8 sub-licenses whilst 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 projects 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 9 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 the 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 is used to give approval to the operator between each key milestone.

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. Experiences in the Co-operative Programme have indicated that the role of robots will be considerably less than expected in the earlier days. 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 them on a full scale (cold) model of one of the major cells. The facility was controlled and monitored through a large 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 where 2 robotic arms (one purchased from Norson Power, UK 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 never used. The second robot is only partially used in defined operations where there are back up solutions available replacing the robot.

Instead of concentrating on totally remote/robotic methods, the approach seems to be developing:

- To use long handled tools with shielding
- To create a less hostile environment by identifying and removing the high sources of radiation as early as possible.

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
- Developed the CODRO (Contact Deployment Remote Operation) approach, which is less expensive and avoids many of the difficulties encountered in a totally robotic operation.

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 seepage of contamination into the cracks caused by penetrations in the concrete of cell walls. Several cm of concrete sometimes must be removed before the surface can be declared “clean”.
- Reappearance of activity on walls previously declared “clean”.

Industrial re-organisation

The time schedule 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

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. 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 this 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. Numbers taken at face value, without regard to their context, are easily misunderstood and misinterpreted.

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 [1].

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 these concurrent activities and their similar aims, and on the initiative of the Co-operative Programme, the three organisations agreed to start a co-ordinated action in order to produce a standardised or uniform listing of cost items and related cost-item definitions for decommissioning projects. Such a standardised list as described previously, 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.

Work on the co-ordinated action has resulted in a new uniform and complete approach to decommissioning costs, which has been presented in an interim technical report [2]. Input to the report was developed by experts representing the three organisations and their supporting groups, including some non-nuclear organisations. The listing produced by the co-ordinated action will therefore probably be utilised by a number of non-nuclear industries.

It is recognised that at this stage the listing in the report has achieved approval in theory but should be further evaluated in practice. This is the reason why it is proposed that this list should be viewed as an interim version, to be broadly distributed, discussed and used, and should be revisited, most effectively in a workshop format, after approximately three years. At that point, a more definitive and more broadly tested and supported report should be issued.

The Task Group has prepared a questionnaire based on the standardised cost item list as well as a manual to help projects to complete the questionnaire. Currently the questionnaires are with the projects in the process of completion.

It should be appropriate to note that the nuclear industry has been the first industry to plan and cost estimate the dismantling and waste disposal of its production plant at the end of its useful life. An example of this is the request by the United States Securities and Exchange Commission to the Financial Accounts Standards Board to develop guidance on unfunded liability of decommissioning costs for industrial plants, including oil and gas production rigs, mines, hazardous material storage facilities, etc.

3.3.2 Task Group on Recycling and Reuse

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. The recycling of such material (or its reuse or disposal) without radiological restrictions could be a significant means of achieving the aim of waste minimisation. So, in 1992, the Programme set up a task group to study the recycling and reuse 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 Co-operative Programme’s Task Group on Recycling and Reuse 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. A report of the work of the Task Group was published in 1996 [3].

Other international organisations like the IAEA and the EC recommendations consider only the radiological risks associated with the release of material, the individual risk corresponding to that resulting from exposure to a maximum of 10 µSv/year. The Task Group assessed the total health risks, comparing 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 show that:

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

3.3.3 Task Group on Decontamination

In its thirteenth meeting, October 20-23, 1992, in Rome, 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 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.

Through this process a list of decontamination processes has been 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.

The objective of this overview of decontamination techniques was to describe critical elements to be considered when selecting techniques for a practical decontamination problem.

The Task Group on Decontamination asked its members as well as other experts to complete individual questionnaires for specific applications. Considerable delays occurred in obtaining completed questionnaires. This delay was mainly due to the difficulties encountered in motivating the Task Group’s contact persons to collect and to transfer the required information. When it became clear that not all questionnaires and relevant information would be made available, it was decided to end the work of the Task Group, and to limit the inventory to the data obtained at that time. Based on this information, a draft final report of the Task Group on Decontamination was prepared by the Chairman of the group with the aim to produce a general overview document and an appendix with the technical details.

The draft was revised by the Task Group members, the TAG and the LC before a final version was approved by the OECD/NEA [4].

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

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 Reuse, 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 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 final draft of the Task Group's report, except for the conclusions, was presented to the LC (today called Management Board) in October 1999.

References

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4. SIGNIFICANCE OF THE PROGRAMME FOR THE PARTICIPANTS

As mentioned earlier in the report, 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 with the same participating member.

So the knowledge and information gleaned from the Programme is both generally applicable and of common interest at one level and is 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.

The main forum of information exchange between the projects has been, as stated earlier, the TAG meetings. Another important functional area of the Co-operative Programme has been the work in the various task groups, particularly the Task Group on Recycling and Reuse and the Task Group on Decommissioning Costs. The results of the work of the task groups have, at a drafting stage, been discussed and analysed at meetings of the TAG. Among other examples of discussions at the TAG which have helped in making project decisions or influencing general project directions are listed below:

- 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 metal material after decontamination could be identified as a technique to characterise the ingots for unconditional release.
- 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 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 to enable their use in a nuclear environment with the required reliability and the required safety, in order to increase the comfort for the operators when compared to working with manually operated tools, but keeping overall control with the operator. Based on a good co-operation with the non-nuclear industry, excellent results may be obtained.

- 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.

- 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,
- The development of microwaves for concrete decontamination machines,
- The development of 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.

- 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.
- 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 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.
- 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.

- 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.

Decommissioning of nuclear installations will become, in a few decades one of the important sectors in the nuclear market probably being of similar importance to the design, construction and operation of nuclear installations. In any case the lessons learned during decommissioning have to be made available in the nuclear field in order to improve design, construction and operation of future nuclear installations.

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. This 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 their nuclear installations.

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

In addition to the tangible benefits listed above is the 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

The OECD Nuclear Energy Agency’s Co-operative Programme on Decommissioning was initiated on the basis of a proposal from the USDOE in 1985. Its main purpose was and still is the exchange of technical and scientific information arising from the planning and execution of major decommissioning projects on nuclear facilities. Starting with the modest number of 10 projects from 7 OECD members, it has grown to be the major forum in the world for this purpose. This is clearly reflected by the fact that by the end of 2002 the Programme has 41 projects from 12 participating members together with several new projects knocking on the door for admission.

Some of the possible reasons for this development:

- Participation is purely voluntary, based on the principle of give-and-take.
- The TAG meetings have become a unique forum for free, open and frank discussions in depth as well as comparisons of practical shop-floor problems and approaches in nuclear decommissioning.
- At the same time, 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.

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.

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

The organisation of the TAG meetings may have to be modified to adjust to the larger number of participating projects, but, as can be observed at present, the current procedures are not yet at a point of saturation. The organisation of the Co-operative Programme as a whole can also be expected to evolve to meet the requirements that arise. In addition, in 2004 the Programme agreed to continue, entering its fourth five-year mandate. With this, the formal CPD Agreement, which is a signed, legal document describing the programme and responsibilities and commitments of its participants, has been updated. This updating includes new provisions for the shared financing of the Programme, demonstrating the continued and increasing value of this work to its participants. Looking back over the twenty years since the OECD Nuclear Energy Agency established the Co-operative Programme on Decommissioning; the Programme has functioned as the main international forum for the exchange of technical and other information arising from nuclear decommissioning projects.

During these past 20 years, nuclear decommissioning has grown from local specialist activities to a competitive commercial industry. In specific areas of common interest, the results of the various task groups of the Co-operative Programme are being used to further the interests of nuclear decommissioning and of the nuclear power industry as a whole.

The WPDD is thankful to the CPD for bringing forward this report and will be happy to publish it to exchange the information in a wider circle. WPDD would also like to encourage other decommissioning projects to join the CPD and the benefitial information exchange within this joint programme.

Annex 1

DESCRIPTION OF PARTICIPATING PROJECTS

The projects participating in the Co-operative Programme are described in this Annex, under the following headings:

A1.1 Completed Reactor Projects, i.e. decommissioned to Stage 3 or placed in a “dormancy” status (Stage 2 or Stage 1), (11 projects).

A1.2 Reactor Projects in Progress, (19 projects).

A1.3 Completed Fuel Facility Projects, (2 projects).

A1.4 Fuel Facility Projects in Progress. (16 projects).

A1.1 Completed Reactor Projects (As of December 2003)

A1.1.1 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 cost 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).

A1.1.2 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 1990’s 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.

A1.1.3 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.

A1.1.4 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.

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.

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 SOCODIE.

A1.1.5 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 or
- 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 per cent verification measurements by the inspection authority and some more (2-3 per cent) 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.

A1.1.6 Kernkraftwerk Lingen (KWL), Germany

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 per cent 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.

A1.1.7 Heissdampfreaktor HDR, Germany

The Heissdampfreaktor (HDR) was a 100 MWt, nuclear superheat 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 $2.2\text{x}10^{10}$ 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.

A1.1.8 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 underwater to the spent fuel storage pool through the canal. These pieces were cut into smaller segments suitable for packaging using another underwater 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 underwater arc saw cutting system. Before assembling the underwater 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 underwater.

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 rad-waste 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), and
- 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 sec 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.

A1.1.9 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 2246.8 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. It is worthy of note, however 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.

A1.1.10 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.

A1.1.11 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-per-cent capacity factor/<30-per-cent 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 cent 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 per cent 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.

A1.2 Reactor Projects in Progress

A1.2.1 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 underwater, 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, Pressurizer, 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 be 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.

A1.2.2 Paldiski, Estonia (This Project has ceased being a member of 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 (1.2x1.2x1.2m) 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 walls; 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 1m³ concrete waste containers, 3 control rod containers, 8 steam generators, and 67x200 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 m³ 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.

A1.2.3 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.

A1.2.4 Bugey 1, France

Bugey 1 was a 540Mwe 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, 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.

A1.2.5 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 15m, height 9m, pond walls thickness 0.8m). 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, CEA decided in 1995 to stop all nuclear activities by 2015 and was faced to the denuclearization 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 fuel was removed, some experimental devices were treated and circuits rinsed, then the facility was kept under surveillance from 1993 to 1999 due to budget constraints.

In the facility some experimental devices were accumulated during its lifetime and wer 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 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 commenced 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 & 3

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

Dosimetry forecast

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

Up to mid 2004

All the wastes produced 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

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 1500 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:	$5x10^{-3}$ GBq

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

A1.2.6 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 is 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 ton steam generators, the 20 t pressurizer 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 and is proceeding. The equipment for dry dismantling of the RPV-components (Lid, upper spacer, lower spacer, RPV) has been installed. The band saw has been tested in the installed state at MZFR. It is planned to start 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 already 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 will be completed in 2003. After that, the measures of the eighth step will start: 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.

The eighth step is in its licensing procedure.

A1.2.7 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 1973-90. 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, 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 will have to be transferred out of the wet on-site storage into dry storage in CASTOR casks by June 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 will be tested on the reactors 7 and 8 which were not in operation (non-activated) in the controlled area of Unit 5, which had only been in operation a short while. Full scale testing of this equipment is currently proceeding.
- 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). 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 reuse projects are proceeding.

A1.2.8 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 licenced 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.

A1.2.9 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 with the dismantling of the primary cell and sodium-cleaning cell as main activities. 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 will cover 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 2003.

KNK is owned by FzK, but is operated by the company KBG, which is a subsidiary of the local utilities. The staff of the KNK has been reduced from 120 persons during operation to 35 (plus 50 contractors' personnel) today, reducing the operational cost by a factor of five.

A1.2.10 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 litre) of super-compacted Dry Active Waste (DAWs), previously stored in 2.429 drums (220 litre), 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 x 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 on January 26th, 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 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 characterization.
- 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.

A1.2.11 Latina, Italy

Latina was a 210 MWe[1] 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.

1. 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.

- 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.

A1.2.12 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.

A1.2.13 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 Greenfield status and will be reused 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 auxilliary 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 has 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 in-active 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).

A1.2.14 Korea Research Reactors 1 & 2 (KRR-1 & 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 3 following categorises: 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.

A1.2.15 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 second of which (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 1.7x1.7x1.7 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^{o}C before being fed into the thin film evaporator, to which bitumen is fed at 110^{o}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.

A1.2.16 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 a level 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. 58 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.

A1.2.17 Taiwan Research Reactor (TRR), Chinese Taipai

The Taiwan Research Reactor at Lung-Tan, Tao-Yuan, (about 200 km south of Taipeh) 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$).

A1.2.18 Windscale Advanced Gas-cooled Reactor (WAGR), United Kingdom

The Windscale Advanced Gas-cooled Reactor was a 100-MWt 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).

A1.2.19 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 2 years.

One of the main activities in the decommissioning project (SDP) is the disposal of both the primary and secondary sodium. For this a Sodium Disposal Plant 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; 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 1000 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.

A1.3 Completed Fuel Facility Projects

A1.3.1 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.

A1.3.2 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)

A1.4. FUEL FACILITY PROJECTS IN PROGRESS

A1.4.1 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 40x40x20 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.

A1.4.2 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 shut down 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.

<u>204A Bays</u>

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

<u>204B Bays</u>

Water	400 m^3	3.4 x 10^{11} Bq	75% ^{137}Cs, 20% ^{90}Sr, 5% other
Algae/sludge, etc.	9 m^3	9.0 x 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.

A1.4.3 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 are no release levels (activity concentration levels) 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 about a 10 cm depth of the floor, 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 $5{,}65 \times 10^{-2}$ Bq/g (alpha) and $4{,}87 \times 10^{-2}$ 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 are expected to be completed by the summer of 2001. The residual contamination on site should be less than 37×10^{6} Bq (1mCi), which is the threshold for Installations Classified for Environmental Protection.

A1.4.4 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 x 78 m). This building is divided into four fire zones making up units. Each unit has an area of 2 000 m^2 (40 x 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 will go on till 2005. The main dismantling operations starting in 2005 will 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 Buildings18, 91 & 54: 84%.

Aspects of technical interest in the project are:

- A CeIV 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).

A1.4.5 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.

A1.4.6 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 COGEMA. There is an agreement between COGEMA 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 characterization.

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.

A1.4.7 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 a 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 450 M€.

A1.4.8 UP1, Marcoule, France

UP1 (Usine Plutonium 1) was an industrialreprocessing 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 gloveboxes.
- very varied nuclear spectrum.
- various radiological incidents during operation.
- large quantity of operational waste.

The decommissioning strategy will utilize 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 2 000 M€ (3 000 M€ including support facilities: laboratories, liquids and solids waste treatment).

A1.4.9 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 workstations, 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. Currently, step 3 is being executed, the aim of which is to “free” all controlled areas in the process building.

Step 3 is being carried out under a series of licences, the first two of which have been granted, the third has been applied for and the application for the fourth is under preparation.

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 is expected to be completed by June 2005. Meanwhile, the plant is being constructed (by FzK,) for vitrification of the high active waste concentrates between 2004 and 2005.

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.

A1.4.10 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 cutout 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.

A1.4.11 Plutonium Fuel Fabrication Facility, Japan

Japan Nuclear Cycle Development Institute has three 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): stabilization 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): Reuse 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.

A1.4.12 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 is to decontaminate stainless steel equipment to reduce radwaste and to decontaminate the building walls to levels for unrestricted reuse. 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 summarized 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.

A1.4.13 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, pressurized 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 characterization 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 65 MSEK. 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 (approx. four 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.

A1.4.14 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 per cent of estimated cost and 30 per cent 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 finalizing 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 reorganized. Later BNF (Inc) had purchased Westinghouse (100% of the electrical part and 60% of the Government and Environmental Departments). This has led to further reorganization.

A1.4.15 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.

A1.4.16 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.

90 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.

Annex 2

REPORT FROM THE TASK GROUP ON DECOMMISSIONING COSTS

A2.1 Introduction

During the second five-year period of the Co-operative Programme, in several projects to decommission various types of nuclear facilities, it was shown that technical methods and equipment are available to safely dismantle nuclear facilities, of whatever type or size. Much experience in the use of these techniques resulted from maintenance and repair work, and from the decommissioning of prototype, demonstration, and small power reactors or other facilities from the nuclear fuel cycle.

The decommission projects also demonstrated that decommissioning costs could be managed. Comparisons of individual cost estimates for specific facilities could show relatively large variations, however, and several studies attempted to identify the reasons for these variations.

In the past, the basis of the cost estimates for decommissioning projects lay in the world-wide experience obtained either in decommissioning projects or in maintenance and repair work at operating nuclear facilities where conditions are to some extent similar. This experience was utilised directly or as an analogue for estimating the costs of similar tasks in decommissioning projects, or indirectly for the assessment of unit costs for basic decontamination and dismantling activities.

Different costing methods have different data requirements, however, and consequently, their reliability depends on the extent to which various data are available and applicable to the specific case being considered. Independent of the assessment method, some uncertainty is inevitable in all estimates of future costs, and no costing method is generally superior to others in this respect. However, analysis of the costing method may be useful in order to locate the key uncertainties in each specific estimate.

As a result, in 1989, the Liaison Committee of the Co-operative Programme for the Exchange of Scientific and Technical Information Concerning Nuclear Installation Decommissioning Projects set up a Task Group on Decommissioning Costs in order to identify reasons for the large variations in reported cost estimates of 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 this matrix.

Cost data was progressively refined by discussions between Task Group and project managers to improve the basis of comparison and to make the data more uniform. Real project specific discrepancies were identified and analysed without bias resulting from inconsistent or inappropriate data.

In addition, the Task Group reviewed some general factors identifying issues dealing with political/geographical, technical and economic/financial aspects causing variations in estimated costs. These factors were only treated qualitatively, since data could not be separated to analyse their quantitative effects.

One of the lessons learned 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. Numbers taken at face value, without regard to their context, are easily misunderstood and misinterpreted.

Another important observation the Task Group made is that there was no standardised listing of cost items or estimating methodology established for 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 such a standardisation [1].

A2.2 Second Task Group on Decommissioning Costs within the Co-operative Programme on Decommissioning

In 1994, the Liaison Committee of the Co-operative Programme decided to re-start the work of the Task Group on Decommissioning Costs. The terms of reference/programme of work for the new study were decided as follows:

- Structure/break down the costs in cost groups/cost items/cost factors; clearly define the scope of each of these, compare the results with other lists (from current studies), and prepare a new "standardised" list.
- Compare/contrast/explain differences in results presented in various countries/projects, looking specifically to commercial nuclear facilities/projects in or related to the Co-operative Programme, separating reactors and fuel facilities in two groups.
- Prepare a questionnaire, and ask participating organisations to provide their relevant cost figures in the standardised list, producing a new inventory of cost estimates (at least six reprocessing plants and over a dozen reactors of various sizes and types, including commercially operated plants, are involved, and also other organisations have shown interest in a co-operation).
- Analyse and scrutinise the cost inventory in order to identify aspects of discrepancy and the reasons for these.

In its early meetings, the Task Group reviewed the list of cost items proposed by the former Task Group. Definitions (a library) at cost item and/or sub-item level were prepared, including a description of the technical activities considered in each cost item. In addition a questionnaire for collecting data from various projects as well as a manual for completing the questionnaire was prepared.

In a later phase, the list of cost items was adapted and was made completely similar to a list of cost items proposed by an IAEA Consultants Group on Decommissioning and Waste Management Costs (see section 3, Activities carried out in other international organisations). In addition, it was decided to start a co-ordinated action with the OECD/NEA, the IAEA and the EC to have the organisations in line to adopt a similar (standardised) list of decommissioning cost items. The three organisations could then further use this standardised list for their own objectives and scope of work, but, at least, in this way would talk the same language. At the same time, results or conclusions of individual evaluation work of different groups would more likely to be consistent instead of being contradictory.

As a result of the decision to start the co-ordinated action with EC, IAEA, and OECD/NEA, the Task Group also decided to adapt its schedule of work, waiting for preliminary approval on the list of cost items and cost item definitions within the international co-operation.

A2.3 Activities carried out in other international organisations

Quite early in the work of the re-started Task Group on Decommissioning Costs, it was noted that activities were going on in other international organisations.

A2.3.1 Activities carried out within the International Atomic Energy Agency (IAEA)

In its 1995-1996 programme, the IAEA initiated a technical document on cost of radioactive waste management and decommissioning with the aim to create a comprehensive list of cost groups, cost elements and cost factors (factors that influence costs) related to waste management and decommissioning from a waste generator/owner point of view.

It was considered to be beneficial to establish a "standard" glossary, providing definitions of technical and cost terms and cost items. It was expected that such a list would facilitate communication, and possibly, encourage common usage among Member States.

A Consultants Group agreed on the definitions of a "cost group", "group of tasks", "cost element", "cost factor", and "cost breakdown units" as "labour cost", "plant & capital equipment" and "expenses". In addition, a list of cost groups and cost items (defined by activities/steps) was defined for both radioactive waste management and decommissioning, being very similar to the list prepared by the Task Group on Decommissioning Costs of the OECD/NEA Co-operative Programme [2]. Further activities were planned in order:

To prepare definitions of the technical cost groups, cost elements, and cost factors;

- To prepare a questionnaire, send it out to volunteer organisations from Member States, review responses, analyse data for consistency and determine if additional clarification is required.
- To edit a technical document, with an introduction, analysis of collected data, and case studies.

A2.3.2 Activities carried out within the European Commission (EC)

In the 1994-1998 Nuclear Fission Safety Programme (section C.4, "Decommissioning of nuclear installations"), the European Commission decided to continue the development of a database on decommissioning costs of nuclear installations, as well as research and development in the field of the dismantling of nuclear installations, particularly relating to issues of environmentally compatible conditioning of radioactive dismantling wastes, the minimisation of radiological impact and the reduction of costs, i.e. by the application of innovative techniques [3, 4].

Objectives of the programme were to develop relevant methodology, to collect, analyse and qualify relevant decommissioning data, to identify, test and evaluate strategic planning tools, and to stimulate the exchange of experience from the decommissioning of nuclear installations.

The existing EC database on costs, occupational doses, waste arising from decommissioning, was set up with the co-operation of various partners within the European Union and was intended to be upgraded in an Oracle 7 environment, enabling "Windows"-like concepts, that are easier to apply by PC-users.

A2.4 Initiation of a co-ordinated action to develop standardised decommissioning cost items

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. It was agreed that the three organisations had very similar objectives with respect to cost items for decommissioning operations, i.e., to facilitate communication, to promote uniformity, to encourage common usage, to avoid inconsistency or contradiction of results or conclusions of cost evaluations, and to be of interest to all those who decommission.

Representatives from the three organisations agreed that a common final document, including a standardised list of cost items and cost item definitions, should be published and that the co-operation could be concluded by organising a common seminar or workshop, where the results of the work could be presented, discussed and demonstrated.

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 and was, as to the structure, very close to the work carried out in the IAEA Consultants Group on Decommissioning and Waste Management Costs, the work of the OECD/NEA Task Group on Decommissioning Costs was used as the basis for further discussions.

Letters of intent were exchanged between the OECD/NEA, the IAEA and the EC and the co-ordinated action to develop a standardised listing of cost items for decommissioning projects with related cost item definitions was officially started in January, 1997. The contribution of the European Commission was incorporated within the EC 1994-1998 Nuclear Fission Safety Programme.

A2.5 Development of a standardised list of decommissioning cost items and their definitions

To achieve the objectives of the co-ordinated action, it was necessary to identify, to define, to harmonise and to verify the general and specific activities carried out during the decommissioning of nuclear facilities, as well as their relating cost items, in order to include these in a standardised list of cost items for decommissioning projects.

A2.5.1 Identification of decommissioning activities and related cost items

The general and specific decommissioning activities and relating cost items considered in the evaluations or specific projects carried out by the individual participating organisations were identified and listed.

The collected information was incorporated into one single and uniform list. Specific meetings were organised, and extended information was exchanged by letter in order to discuss this list with representatives of the three organisations with a view to achieving harmonisation and completeness. After discussion, it was considered that the resulting list was a good basis for a single, uniform and agreed reference list of decommissioning cost items for which specific definitions had to be prepared.

After harmonisation of the collected information, an overall concept was developed to class the decommissioning cost items of the reference list into groups. The concept was based on the approaches adopted within the OECD/NEA Task Group on Decommissioning Costs and the IAEA Consultants Group, in which cost items were grouped that are related to activities that are carried out with a similar emphasis, whether or not tied to a similar time schedule for decommissioning, or that are based on overall activities that cannot be categorised in a specific time period.

Based on these considerations, eleven cost groups were identified:

- Pre-decommissioning actions.
- Facility shutdown activities.
- Procurement of general equipment and material.
- Dismantling activities.
- Waste treatment and disposal.
- Security, surveillance and maintenance.
- Site clean-up and landscaping;
- Project management, engineering and site support.
- Research and development.
- Fuel.
- Other costs.

This list of cost groups was discussed and adopted by the representatives of the three co-operating organisations.

A2.5.2 Identification of definitions of decommissioning cost items

As a next step, the identification and listing of the definitions of the general and specific decommissioning activities and relating cost items considered in the evaluations or specific projects carried out by the individual participating organisations was started.

The information received from the individual organisations was evaluated, compared and compiled into one document in order to present a draft for a single and standardised list of cost items, cost groups and cost item definitions. As indicated in section 5.1, Identification of decommissioning activities and related cost items, the concept was based on the approaches adopted within the OECD/NEA Task Group on Decommissioning Costs and the IAEA Consultants Group, and the principles were not in contradiction to the approach adopted in the EC database on cost.

As a result, definitions for the cost items in the standardised list were prepared considering that:

- Decommissioning activities include an inventory of a coherent set of tasks, that cover the specific aspects that may have to be dealt with during the decommissioning of a nuclear facility, whether or not a specific task will be executed in a specific decommissioning project.

- Processes or work packages comprise a selection of a coherent set of decommissioning activities or tasks that must be carried out as a part of a decommissioning project or as a decommissioning project itself.

  A global decommissioning project with a specific cost comprises a selection of processes or work packages, being as such a collection of dedicated decommissioning activities grouped in specific processes/work packages, that may be universally and independently selected from the standardised list of decommissioning cost items based on the specific application defined in the project itself.

Based on the comments and additional considerations that were received, subdivision of cost item definitions into sub-items was required in order to enable specific identification and comparison of the available information. As a result, a new version of the proposed standardised list of decommissioning cost items and related cost item definitions was prepared, including the comments and considerations received, except for the ones that were not in harmony with the general concepts described in the foregoing sections.

A2.5.3 Identification, definition, harmonisation of cost categories for decommissioning activities

In the evaluations or specific projects carried out by the individual participating organisations, the costs resources for the general and specific decommissioning activities and relating cost items are mostly divided in cost categories. A cost category specifies the nature of the cost (e.g. depreciation costs, salary costs, building rent, etc.), and related cost categories may be grouped. The identification and listing of these cost categories and their specific definitions were also completed.

The information received was compared and compiled into one list, presenting a draft for a single and a standardised list of cost categories and related definitions, similarly to what was done for the decommissioning cost item definitions.

A2.5.4 Final report on proposed standard list of decommissioning cost items

A final report 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 common document from the three participating organisations [5].

It is a comprehensive document trying to give a first answer to the detailed comments, questions and remarks received during the years the co-ordinated action lasted, and containing underlying principles reviewed for consistency by the participating organisations.

Although it is hoped that the standardised list will be widely accepted and used, it was recognised that at this stage the list has 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.

A2.6 Continuation of the activities within the Task Group on Decommissioning Costs

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. Some considerations were given to the future work of the Group. It was discussed that further activities should include:

- To finalise and send out the questionnaire and the manual for completing the data (Excel 4.0 format on diskette).
- To request for answers to the questionnaire.
- To evaluate the answers to the questionnaire and to ask for additional information if required.

- To start analysing the collected data if an adequate number of projects respond to the enquiry, i.e. at least 5 questionnaires per group to be evaluated (nuclear power plants, fuel facilities).
- To organise a first evaluation meeting preferably connected to a future TAG-meeting.

As the work involved in these activities was considered to be non negligible, i.e. for the Task Group members to evaluate as well as for the projects to deliver the required information, some TAG members thought it was necessary to leave the decision to continue the work of the Task Group to the members of the Liaison Committee, i.e. the managers of the participating projects.

As a result, a progress report was prepared for the Liaison Committee, including a review of the organisation of the Task Group, an overview of the progress of work, the results obtained so far, and the actual status, a section on the co-ordinated action to produce a standardised or uniform listing of cost items for decommissioning projects as well as the cost item definitions, and an overview of the future work of the Task Group as indicated before.

The Liaison Committee was requested to take notice of this progress report, to agree that further work should be continued and that their participating projects intended to provide the data required in the questionnaire in order to enable the implementation of the proposed analyses.

During its meeting at the end of October 1999, the Liaison Committee discussed the contents of the progress report, and the LC Members agreed to instruct their TAG Members to use the agreed-upon cost structure, as developed in the interim technical report, as well as to provide cost data to the Task Group on Decommissioning Costs for analyses.

As a result of this LC decision the activities in view of finalising the questionnaire and preparing the guidelines for completing the questionnaire were continued.

A2.7 The decommissioning cost questionnaire

An overview document was prepared considering the information required in order to enable development of the proposed analyses, as well as some explanation on how to handle the questionnaire. The overview considered the information and questions comprised in the worksheets of an Excel 4.0 file. A first part requested for some general information about a project. A second part comprised remarks, comments, and questions to be reminded or to be mentioned/considered when completing the questionnaire, while a third part included the actual questionnaire in view of the data collection.

The overview document, the questionnaire as well as the manual were mailed to the members of the Task Group and the representatives in the Technical Advisory Group in May, 2000. As a long time passed since the last detailed discussions about the subject, a short introduction as well as the report from the last meeting of the Task Group was added. As a result, experienced as well as new members of the Technical Advisory Group could familiarise themselves (again) with the items. In addition, at the TAG-meeting in June, 2000 in Knoxville, a demonstration was given on how to complete the questionnaire forms.

A2.8 Expected developments

After distribution of the questionnaire, it was expected that further activities would include:

- To receive the answers to the questionnaire.

- To evaluate the answers to the questionnaire and ask for additional information if required.
- To start the analysis of the collected data if an adequate number of projects respond to the enquiry, i.e., at least 5 questionnaires per group to be evaluated (nuclear power plants, fuel facilities).
- To organise one or more evaluation meetings as required, preferably connected to future TAG-meetings;
- To discuss the results of the analyses and prepare a report with conclusions that responds to the requirements developed in the terms of reference of the Task Group.

It was agreed that completion would require some time. It was proposed, therefore, that questionnaires should be completed by the end of October 2000. A first discussion about the responses was scheduled for the October, 2000 TAG meeting in France.

A2.9 Status overview

At the end of April 2002, it had to be recognized that insufficient data could be obtained in order to enable acceptable evaluations, though in subsequent meetings of the Technical Advisory Group, TAG members were encouraged to provide the required data. As a result, the Technical Advisory Group as well as the Liaison Committee were invited to agree that the activities of the Task Group on Decommissioning Costs were terminated.

Though ultimately it is hoped that the standardised list will achieve wide acceptance and use, it was recognised that until now the list had achieved approval in theory but should be further evaluated in practice. It was therefore proposed that the list should be broadly distributed, discussed and used. The Task Group on Decommissioning Costs of the Co-operative Programme has not been able to achieve this goal, however. Collecting sufficient adequate data proved to be the hampering factor.

Organisation of the Task Group

Teunkens, L.	Belgoprocess	Belgium	Chairman
Millen, D.	Belgoprocess	Belgium	Technical Secretary
Campani, M.	EDF	France	
Jeanjacques, M.	CEA	France	
Nokhamzon, J.G.	CEA	France	
Harbecke, W.	KWL Lingen	Germany	
Yanagihara, S.	JAERI	Japan	
Pettersson, S.	SKB	Sweden	
Lee, J.	Magnox Electric	UK	
Skokan, B.	US DOE	US	
Cloutier, B.	TLG Services	US	
LaGuardia	TLG Services	US	
Abreu, A.	ENRESA	Spain	
Menon, S.	Menon Consulting	Sweden	Programme Co-ordinator

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[5] EUROPEAN COMMISSION, INTERNATIONAL ATOMIC ENERGY AGENCY, NUCLEAR ENERGY AGENCY OF THE ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT, A Proposed Standardised List of Items for Costing Purposes in the Decommissioning of Nuclear Facilities, Interim Technical Document, Paris, 1999.

Annex 3

REPORT FROM THE TASK GROUP ON RECYCLING AND REUSE OF SCRAP METALS - OVERVIEW AND EVALUATION OF COLLECTED LITERATURE REGARDING TECHNICALLY ENHANCED NATURALLY OCCURING RADIOACTIVE MATERIAL (TENORM)

A3.1 Introduction

Since the publication of the report of the Task Group on Recycling and Reuse in 1996 [1], the activities of the Task Group, with the Technical Secretary in particular, have more or less been concentrated on promulgating the views expressed in the report and on the issue of technologically enhanced naturally occurring radioactive material (TENORM) that has emerged during the last decade, which can have a very significant impact on clearance regulations.

Studies have shown that TENORM can be of the same activity levels as low level waste and is very similar to the candidate material for exemption and clearance in the nuclear industry, but occurs in many non-nuclear industries in quantities that are huge in comparison (2-3 orders of magnitude larger quantities than those used in European studies on nuclear recycling). The emergence of the issue of TENORM, with its huge quantities, its activity levels and the large number of industries involved, should help the nuclear power industry to place the issue of ionising radiation in perspective.

Among the basic aspects that were noted regarding the regulation of low dose radiating materials were the following:

Comparison of risks between nuclear and non-nuclear radioactivity

Different dose criteria are being used for evaluating risks for treating radioactive material from the nuclear industry and that arising from the non-nuclear NORM industries. In this connection the US National Academy of Sciences has clearly rejected such differing treatment. In its "Evaluation of EPA Guidelines for Exposure to NORM [2]", it states:

"The committee is not aware of any evidence that the properties of NORM differ from the properties of any other radionuclides in ways that would necessitate the development of different approaches to risk assessment. In regard to radiological properties, if one accepts the view currently held by all regulatory and advisory organisations involved in radiation protection that estimates of absorbed dose in tissue are the fundamental physical quantities that determine radiation risks for any exposure situation, there is no plausible rationale for any differences in risks due to ionising radiation arising from naturally occurring and any other radionuclides, because absorbed dose in tissue depends only on the radiation type and its energy, not on the source of the radiation".

High background dose radiation areas

It was observed that tens of thousands of people living in areas of high background doses could be suitably studied from the point of view of radiation protection of the public. An example is the population of Ramsar.

Ramsar is a city on the Caspian Sea in northern Iran. The 2000 inhabitants of this city receive an annual absorbed dose from external beta-gamma radiation alone of up to 260 mSv/year, which is many times higher than the 20 mSv/year, that is the permitted dose for workers at many nuclear power stations. The high radiation levels are due to the presence of ^{226}Ra in the local rocks, which are used in the building of most of the houses in the city.

A presentation was made at the recent VALDOR (VALues in Decisions On Risks) international conference on the results of some preliminary biological studies on the citizens of Ramsar [3].

In addition to the external beta-gamma radiation, the inhabitants are exposed to ground water radium concentrations of several hundred Bq/*l* plus the radium in the food, as well as indoor radon concentrations of up to several thousand Bq/m^3. The inhabitants of Ramsar have thus been subjected to a wide range of exposure levels and types of exposure (external beta-gamma, inhaled radon, ingested radium) over several generations. Thus they appear to constitute an appropriate group for being the basis for the formulation of radiation protection measures for the public.

The results of the preliminary biological studies show that:

- Cancer mortality and life expectancy do not appear to be different in the High Background Radiation Areas (HBRA) and in near-by Normal Background Radiation Areas (NBRA). These results are at present based on anecdotal information and an epidemiological study has been started to confirm them.
- Citogenic tests have shown that there are no statistically significant differences between HBRA and NBRA residents. Other testing has shown that there is no reduction in immune system functions or adverse haematological effects among Ramsar citizens compared with NBRA residents.
- The most interesting results were those of an in vitro exposure of blood samples (lymphocytes) from people from both HBRA and NBRA to a “challenge” dose of 1.5 Gy of gamma radiation. Here, the HBRA residents showed only 56% of the average number of induced chromosomal abnormalities of NBRA inhabitants, indicating the development of certain adaptive response to radiation dose in the HBRA residents.

The authors note that similar studies at other HBRAs such as Yangjiang, China and Kerala, India, had also given similar results regarding cancer mortality, life expectancy, chromosome aberrations and immune function.

A3.2 Overview of procedures for the release of material from the nuclear fuel cycle

In 1988, the International Atomic Energy Agency (IAEA) and the OECD Nuclear Energy Agency (NEA), in co-operation, issued Safety Series No. 89 [4] to recommend a policy for exemptions (i.e. clearance) from the basic safety system of notification, registration and licensing that form the basis of regulatory control.

Safety Series No. 89 suggests:

- A maximum individual dose per practice of about 10 µSv/year.
- A maximum collective dose per practice of 1 man Sv/year.

to determine whether the material can be cleared from regulatory control or other options should be examined. Safety Series No 89 is currently being revised.

A methodology to apply the principles of Safety Series No 89 on the recycling or reuse of material from nuclear facilities was subsequently presented [5]. The results of this document were part of the input in the IAEA process of establishing unconditional release levels for solid materials [6]. This last mentioned report, IAEA TECDOC 855, was issued in January 1996 on an interim basis and is being revised after about three years, to react to comments received and to experience gained in its application. The document recommended nuclide specific clearance levels for solid materials.

EC recommendations - Radiation Protection 89 [7] - were published in 1998 for the recycling of metals from the dismantling of nuclear installations. The proposals cover steel, aluminium, copper and alloys of these metals. While the IAEA TECDOC 855 treated only unconditional clearance, the EC approach provides two options for releasing material:

- Direct release based only on surface contamination.
- Melting at a commercial foundry followed by recycle and reuse; mass specific and surface specific levels are provided.

The nuclide specific clearance levels in Radiation Protection 89 are also based on the Safety Series No. 89 criteria.

Earlier, a revised *International Basic Safety Standards for Protection against Ionising Radiation and the Safety of Radiation Sources (BSS)* had been published in 1994. It was based on the recommendations of ICRP 60 [8] and jointly sponsored by the Food and Agricultural Organisation (FAO), the IAEA, the International Labour Organisation (ILO), the OECD/NEA, the World Health Organisation (WHO) and the Pan American Health Organisation (PAHO). The International BSS gives a list of nuclide specific exemption values (both quantities and concentrations).

The EC issued, in May 1996, a Council Directive laying down its BSS for radiation protection [9], with nuclide specific exemption values very similar to those in the International BSS. However, the EC BSS makes a difference between 'practices' covering processes utilising the radioactive, fissile or fertile properties of natural or artificial radionuclides (i.e., the nuclear industry) and 'work activities' where radioactivity is incidental, but can lead to significant exposure of workers or the public (i.e. the TENORM industries).

The USNRC regulation on radiological criteria for the release of a nuclear site for unrestricted use was published in July 1997 [10]. The individual dose criterion to be used according to this NRC regulation is a maximum of 250 µSv/year to be compared to the 10 µSv/year from Safety Series No 89. The USNRC also published draft criteria NUREG-1640 for the clearance of equipment and material from nuclear facilities in January 1999 [11]. These were, however, based on 10 µSv/year maximum allowable individual doses.

The Health Physics Society has endorsed the ANSI Document N13.12, 'Surface and Volume Radioactivity Standards for Unconditional Releases [12]. This has been suggested as an alternative to the draft NRC criteria NUREG-1640. N13.12 is also based on a 10 µSv/a individual dose criterion, while until a year or so ago, the ANSI N13.12 draft was still based on 100 µSv/a.

A3.3 TENORM Quantities

Radiation protection and the management of radioactive material have hitherto been concerned mainly with artificial nuclides arising within the nuclear fuel cycle. In the last few years, there has been an increasing awareness of naturally occurring radioactive material (NORM), however, and the enhancement of its concentration in various non-nuclear industrial processes. This technologically enhanced naturally occurring radioactive material (TENORM) can be of the same activity levels as low level waste and is very similar to the candidate material for exemption and clearance in the nuclear industry, but occurs in quantities that are huge in comparison.

Table 1 illustrates some of the technologically enhanced NORM arising annually in the United States [13]. ^{226}Ra with a half-life of 1 600 years is by far the most important radionuclide. These data are only shown to give an idea of quantities and activity levels. Other industries with significant radioactive waste streams are petroleum processing, geothermal plants and paper mills. More or less comparable quantities of TENORM arise in Europe, with similar concentrations of radioactivity [14].

The quantities shown above should be viewed in comparison to candidate material for recycling from the nuclear industry. The European studies for recycling of steel from nuclear facilities have used a basis of 10 000 t/year [7]. The OECD/NEA Task Group on Recycling and Reuse used a quantity of 50 000 t/year in the United States in their study [1].

Table 1: Some NORM Quantities [13]

Waste Stream	Production rate (t/year)	U+Th+Ra (Bq/g)
Phosphates	5×10^7	up to 3 700
Coal ash Petroleum production	6.1×10^7 2.6×10^5	up to 2 up to 3 700
Water treatment	3×10^5	up to 1 500
Mineral processing	10^9	up to 1 100

A3.4 TENORM Regulation

A3.4.1 Background

The regulatory structure for exempting or releasing material from radiological control is based on the principle of triviality of individual doses to members of the public. The ICRP criterion of “some tens of microsieverts” became ‘ten microsievert or less’ in the IAEA Safety Series No 89, which was created at a time when TENORM was unknown or, at any rate, not considered. The one and the same criterion was later used for two regulatory concepts: **exemption** (from entering regulation), and **clearance** (for release from regulation), with generally a factor ten higher activity concentration values for exemption as for clearance. The difference in activity levels was explained by ‘quantities’, exemption being applied to small (“moderate”) quantities and clearance to large quantities. In practice, ‘small’ meant say 1-10 t, while in European studies on (clearance for) recycling, the figure of 10 000 t has been used to exemplify ‘large’ quantities.

Later TENORM was discovered. Its quantities (2 to 3 orders of magnitude larger than those used in the European studies on nuclear recycling), its activity levels and the large number of industries involved are being or have been mapped.

A3.4.2 The EC Approach

The European Commission, in their BSS [9], propose to solve this problem by dividing occurrences of radioactivity into:

- *Practices*, which utilise the radioactive properties of materials, i.e. the nuclear industry.
- *Work activities*, where radioactivity is incidental (TENORM industries).

The EC-BSS prescribes an individual dose constraint of 10 µSv/year/practice for the nuclear industry. It gives a nuclide specific table of exemption levels for practices. A typical value for nuclides of interest (^{60}Co, ^{137}Cs, and ^{226}Ra) is 10 Bq/g. The BSS does not give a corresponding table for work activities. However, it was noted at the

NORM II meeting in Krefeld, Germany [15], that much higher levels were being used in certain European countries ranging from 100 – 500 Bq/g.

Later the EC has published document Radiation Protection 122, Part 1 dealing with practices and Part 2 regarding the application of the concepts of Exemption and Clearance to Natural Radiation Sources. The EC suggests a release criterion of 10 µSv/year for material from the nuclear industry in Part 1 and 300 µSv/year in Part 2 for TENORM. It justified the selection of the 300 µSv/year criterion by the following:

- It is comparable to regional variations in dose from natural background radiation.
- It is coherent with exemption levels for building materials (in Radiation Protection 112).
- It is coherent with dose constraints for effluents to air and water (300 µSv recommended by ICRP for the nuclear industry).
- It is below the lower marker point for worker exposure in "work activities" (EC term for non-nuclear industries).

It is to be noted that all the above justifications are equally relevant for the clearance of material from the nuclear industry.

A3.4.3 The IAEA Approach

It seems that the IAEA is considering to propose the 10 µSv/year individual dose criterion for the nuclear industry and 'optimisation' in each individual case of TENORM regulation. The process of optimisation seems vague and undefined. It seems to be 'intuitive' rather than being based on any formal risk and cost/benefit analysis. In the IAEA TECDOC 855, there is reference to the optimisation of radiation protection using "*cost-benefit analysis, intuitive or formal, or other methods*". Another IAEA document, TECDOC 987, has an Appendix II on the justification and optimisation of clean up. The paper refers to "*multi-attribute utility analysis*", and gives an example of an equation, where the net benefit is a function of a number of parameters like avertable collective dose, monetary costs of clean-up, anxiety regarding the contamination, reassurance by the clean-up, etc. It can be stated about such an 'optimisation' that:

- It is arbitrary; the dollar values of the parameters, specially the last two, can be chosen to give any predetermined result.
- Such "optimization" will lead to different results in calculations by different authorities in different states; consistency, harmonisation of regulations as well as trans-boundary transport will be impaired.

- Such calculations will be difficult to explain in communication with the public and difficult to defend in a public debate.

In the summer of 2001, the IAEA presented a basically new approach by suggesting

- It is sensible to use one unique set of radionuclide specific levels for the purpose of indicating a boundary between radioactive material that may not warrant imposition of the regulatory system and material that may warrant regulation.

Preliminary proposals:

- Single set of values for defining scope of BSS in terms of Bq/g (would, in principle, replace previous generic exemption levels, clearance levels and commodity levels).
- Applies to all materials except food and water.
- The BSS would be modified by introducing a definition of its scope and removing existing exemption levels and references to clearance.

These proposals are laid out in a report entitled The Scope of Radiation Protection Safety Standards: Strategy for Rationalisation of Policy.

A3.4.4 Proposed ANSI Guide

The ANSI guide N13.53 for the control and release of TENORM [16] has administrative release levels based on a maximum of 100 µSv/year ('less than 10 mrem in practice'). It does seem rather peculiar that, in spite of the 100 µSv/year, instead of 10 µSv/year as used by IAEA and EC, the release level for ^{210}Po, ^{210}Pb, ^{226}Ra and other nuclides of the Thorium series is only 0.1 Bq/g, compared to the IAEA's 0.1 to 1 Bq/g (with a representative value of 0.3 Bq/g) and EC's 1 Bq/g.

A3.5 Examples of regulation of TENORM

A3.5.1 Recycling and Reuse of TENORM

Slag from steel melting

A European company has melted 350 Mg of scrap [17] from the natural gas industry resulting in:

- 18 Mg of slag with average specific activity: 93 Bq/g.
- 1 Mg of filter dust with average specific activity: 35 Bq/g.
- 3.6 Mg of floor sweepings with average specific activity: 255 Bq/g.

Four of the waste drums exceeded the exemption level of 500 Bq/g. The Federal Collection Depot for radioactive waste offered to store 3 of these for the price of 475 000 DEM. The fourth drum was refused because the activity level of ^{226}Ra was too high.

'Practicable and economic' waste management alternatives were sought and the radiological impact of five such alternatives were studied: road construction, shallow land burial, sidewalk, playground, or parking lot. Using the slag for road construction was finally the chosen method of waste management, and the allowed individual dose criterion was 1 mSv/year.

At the same company, radiologically similar slag arises from the melting of material used in ex-vessel core melt experiments (metals with depleted UO_2 powder added to simulate fuel) and scrap from fuel element fabrication. The slag from these melting operations, being from the nuclear industry, is proposed to be regulated under the 10 µSv/year individual dose criterion.

Coal ash

According to UNSCEAR, 280 million tons of coal ash arise globally every year. 40 million tons are used in the production of bricks and cement and "a great deal" is utilised as road stabiliser, road fill, asphalt mix and fertiliser. Annual doses to residents can be up to several mSv. These doses are presumably only the gamma component. The main radioactive nuclide in most TENORM is ^{226}Ra and, as the IAEA draft report [18] points out, SENES has calculated a dose of around 10 mSv/a from 1 Bq of ^{226}Ra via the indoor radon exposure pathway. So, in addition to the gamma doses, there will also be a considerable dose from the radon.

About 61 million tons of coal ashes were generated in the United States by thermal power production in 1990 [13]. Such ash is either disposed or utilised for various industrial applications (more than half for the production of concrete/cement). About 6 million tons of coal ash, with TENORM, is exempted from regulation by the US Environmental Protection Agency (USEPA) for use in building materials. The resulting individual dose to members of the public can be about 100 µSv/a [19]. The distribution in 1990 between the two alternatives was about 80% disposal to 20% utilisation. The American Coal Ash Association hopes to ultimately reverse this distribution to 20% disposal and 80% utilisation. It is pointed out that such a high utilisation rate is technically achievable, as rates up to 70% utilisation are not uncommon in Europe.

In Europe, every year about 30 million tons of coal ashes are generated. If the American Coal Ash Association is correct, about 21 million tons are being utilised.

A3.5.2 Disposal Aspects of TENORM

The major TENORM radionuclide is ^{226}Ra, with a half-life of 1 600 years, while the dominating nuclides in scrap from the nuclear industry are ^{60}Co (half-life 5.4 years) and ^{137}Cs (half-life 30 years). Current regulations at many near surface repositories have stringent limits on the quantities and concentrations of long-lived nuclides in disposed material, limits that may well make it necessary – according to current regulations for nuclear industry waste – to condemn non-exempted TENORM to deep geological disposal. According to the currently proposed criteria, the same nuclide, at the same concentration, can either be sent to deep geological disposal or release for use in road repair, depending on whether it came from the nuclear industry or a non-nuclear one.

The IAEA has started to study the implications of the need for disposal of huge quantities of such long-lived nuclides. A draft paper has been produced on a common framework for the principles of the management of all radioactive waste, including waste from mining and processing of radioactive ores and minerals [18]. The document does not, however, consider the candidate material for recycling/reuse or utilisation of very low-level radioactive waste. The draft paper mentions mining and milling wastes (MMW) and some other types of slightly radioactive waste streams from non-nuclear industries (TENORM) but does not mention the largest waste stream of this kind: coal ash.

A3.5.3 Commercial Aspects

When decommissioning nuclear facilities, the costs for the management of redundant material and its disposal are very significant. They can be up to 50-60 % of the total decommissioning costs. A major part of this redundant material is very slightly contaminated by radioactivity. The proposed criteria for the release of such material is 30-100 times more stringent for the nuclear power industry than for the competing fossil fired power industries, which gives the latter a significant price advantage.

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Annex 4

REPORT FROM THE TASK GROUP ON RELEASE MEASUREMENTS

A4.1 Introduction

The Task Group on Recycling and Reuse concluded that after treatment significant quantities of waste generated from decommissioning can be recycled and reused. Indeed, recycle and reuse options provide a cost effective solution to the management of waste arisings. The most significant impediment to the use of recycle and reuse is the absence of consistent release standards within the nuclear industry. Several international organisations, for examp[le IAEA and the EC, have proposed standards with the object of agreeing to an internationally accep[ted set of release standards.

The current recommendations of international organisations are aimed solely at minimising radiological risks. The philosophy behind these recommendations are based on IAEA document, Safety Series No. 89 (1-2),which was issued jointly by the IAEA and the OECD Nuclear Energy Agency (NEA) in 1988. Safety Series 89 suggests:

- a maximum individual dose/practice of about 10 µSv/year.
- a maximum collective dose/practice of 1 man Sievert/year,

to determine whether the material can be cleared from regulatory control or other option should be examined.

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 regulation.

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 was therefore established a Task Group to study these problems in an analytical and structured manner.

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Annex 5

REPORT FROM THE TASK GROUP ON DECONTAMINATION

A5.1 Introduction

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.

The Task Group on Decontamination was set up to prepare a state-of-the-art report on decontamination in connection with decommissioning.

The work focussed on decontamination for dose reduction as well as for waste decategorisation or for conditional or unconditional release of materials. The decontamination of both metallic and concrete surfaces was considered. A questionnaire was sent to Project Managers requesting information on the technical and economic aspects of the selected decontamination techniques. In addition to this a separate questionnaire was required for each specific application of a given process, including actual data on the efficiency of the process as well as data on operating and investment costs.

This overview of decontamination techniques is intended to describe some of the critical elements involved in choosing techniques to address practical decontamination problems. The information presented in the Task Group's final report represents a state-of-the-art view of the decontamination techniques available.

A5.2 Selection of Decontamination Techniques for Decommissioning

Very early in the process of selecting decontamination technologies for decommissioning it is important that a cost/benefit analysis is performed 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.

To achieve a good decontamination factor (DF) a decontamination process must be designed for site-specific application taking into account a wide variety of parameters such as:

- type of plant and plant process, reactor type, reprocessing plant etc;
- operating history of plant;
- type of material: steel, zircalloy, concrete etc.;
- type of surface: rough, porous, coated etc.;
- type of contaminant: oxide, crud, sludge, loose etc.;
- composition of the contaminant (i.e. activation products, fission products, actinides etc. and the specific radionuclides involved;

- ease of access to areas/plant to be decontaminated, external or internal surfaces to be cleaned;
- regulatory requirements and the decontamination factor required;
- destination of the components being decontaminated: disposal, reuse etc.;
- time required for application;
- proven efficiency of the process for the type of contamination in the facility;
- type of component: pipe, tank etc.

Other factors are important in the selection process but which do not affect the decontamination factor are:

- availability, cost and complexity of the decontamination equipment and consumables;
- need and capability of treatment and conditioning of the secondary waste generated;
- potential exposure to hazardous materials and/or chemicals used in the decontamination process;
- occupational and public doses resulting from decontamination (justification of the practice);
- other safety, environmental and social issues;
- availability of trained staff;
- extent to which the plant needs to be decontaminated to achieve acceptable conditions for decommissioning;
- salvage value of materials which otherwise be disposed of;
- extent to which the facility must be modified to do the decontamination: isolate systems, enclosed and ventilated spaces etc..

In addition, the choice of a process or of a combination of several processes will finally depend on several other factors such as:

- the specific nature of the application, the complexity of the system.
- the feasibility of industrialization.
- the cost/benefit analysis taking into account all aspects of the decontamination operation, i.e. until disposal of remaining radioactive waste.

The decision whether to proceed with decontamination and the final process selected will depend on the best overall balance of the above factors in order to minimise the overall impact of the decommissioning activities on workers, the public and the environment at acceptable costs.

A5.3 Conclusions

It was the aim of the Task Group to prepare a state-of-art report on decontamination in connection with decommissioning and to describe some critical elements for the selection of appropriate techniques in order to resolve practical decontamination problems. The work has been focussed on decontamination for dose reduction, waste decategorisation and conditional or unconditional release of materials. The decontamination of both metallic and concrete surfaces were considered.

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 were 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.

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