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Radioactive Waste Management Committee

THE NEA CO-OPERATIVE PROGRAMME ON DECOMMISSIONING

TWENTY-FIVE YEARS OF PROGRESS; THE LAST FIVE YEARS - 2006 through 2010

The corresponding RWMC report is also available on the OECD Nuclear Energy Agency webpage and as a printed publication.

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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 among member country organisations actively executing or planning the decommissioning of nuclear facilities. Initiated in 1985, the CPD recently completed 25 years of operation.

The objective of the CPD is to acquire information from operational experience in conducting specific decommissioning projects that is useful for future projects. Its working method is based on the exchange of knowledge currently drawn from 59 specific decommissioning projects. Such information includes, but is not limited to, project descriptions and plans; data obtained from research and development associated with decommissioning projects; and data and lessons learnt resulting from the execution of a decommissioning project.

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

This report, prepared by the CPD, provides a general description of the Co-operative Programme and describes the progress and generic results obtained by the Co-operative Programme on Decommissioning with a focus on the period 2006–2010. It follows similar status reports published by the NEA in 1996 and 2006.[1,2] The RWMC Working Party on Decommissioning and Dismantling (WPDD) has found the information presented by the CPD in this and previous reports valuable for all NEA member countries and therefore decided to publish this report to encourage other member countries and decommissioning projects to consider joining the CPD.

The RWMC and its Working Party on Decommissioning and Dismantling is grateful to the CPD for sharing the experience from its important work.

1. NEA (1996), The NEA Co-operative Programme on Decommissioning: The First Ten Years, 1985-1995, OECD/NEA, Paris.
2. NEA (2006), The NEA Co-operative Programme on Decommissioning: A Decade of Progress, OECD/NEA, Paris.

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Summary

The Co-operative Programme for the Exchange of Scientific and Technical Information Concerning Nuclear Installation Decommissioning (CPD) is a joint undertaking according to Article 5 of the Statute of the NEA. Concluded in 1985, the Agreement constituting the CPD has been continuously extended, although modified in 2003, with the current programme period lasting until the end of 2013. This report provides information about the participants, structure and achievements of the Co-operative Programme and the projects involved.

The objective of the CPD is to acquire information and share operational experience from the conduct of 59 current decommissioning projects, such as project description and design, data resulting from the execution of decommissioning projects, and associated research and development results. The information generated in the project is protected by confidentiality provisions, which allow for a frank and open exchange of experiences, on a "give and take" basis. The information exchange aims to ensure that the best internationally available experience is available and that safe, environmentally friendly and cost effective methods are employed in all decommissioning projects.

The Co-operative Programme is implemented by a Management Board (MB) representing the participating organisations and a Technical Advisory Group (TAG) for the information exchange between the individual decommissioning projects. A Programme Co-ordinator provides secretariat services to the TAG and is the interface to the Management Board and the NEA Secretariat.

The projects in the Programme have a broad range of characteristics and cover various types of reactors and fuel facilities. The number of projects in the programme has grown from 42 to 59 over the past five years. The Programme now covers 35 reactor related projects and 24 fuel related projects representing a wide selection of facility types in each category. Also, all phases of decommissioning – from active dismantling to safe store and to completed decommissioning back to "green field conditions" – are represented.

Over the 25 years of experience of the Co-operative Programme on Decommissioning, and in particular through the information exchange and review within the TAG, it has become evident that:

- decommissioning can and has been done in a safe, cost-effective and environmentally friendly manner;
- the evolution of technologies have demonstrated their effectiveness in performance improvements in all aspects of conducting decommissioning projects;
- the upkeep and maintenance of design, construction and operational records can significantly enhance performance through all stages of a decommissioning project;
- in the absence of waste disposal facilities, interim waste storage facilities with integrated waste processing facilities can effectively be used to keep all levels of waste streams moving and avoid delays to project schedules;
- cleanup of material for recycle and reuse or disposal as conventional waste is cost-effective, environmentally friendly and generally receives positive public opinion;
- prompt decommissioning is increasingly becoming the strategy of choice due to advantages in overall cost and greater public acceptance.

Regarding technical challenges, specific trends have been observed over the last decade. Large contaminated components, for example heat exchangers, steam generators, large tanks etc., that have previously been segmented *in situ* into smaller pieces, are increasingly removed “in one piece” and transported outside the contained area into separated facilities for further processing. Regarding the use of robotics, the CPD observed that industrial robots may have a limited practical applicability in decommissioning contrary to earlier expectations that robotic methods would be extensively used in the decontamination and dismantling of radioactive structures and components although they will remain necessary for some applications especially in the high radiation areas. The cleanup and verification for the release or declassification of alpha contaminated concrete structures where seepage of contamination into cracks and along pipe penetrations has proved to be very challenging and in fact in some cases has prompted authorities to impose much more stringent release criteria.

One key organisational issue that has arisen is the clear need for very tight contract specification in the use of contractors. The scope of the contract has to be very precise and well defined otherwise disagreement and conflict can have serious impact on cost and schedule. Many projects have in fact decided to perform the work in-house where possible using experienced staff and reducing duplication of management oversight effort. The use of a staged licensing process is evident in certain countries. On the surface this may appear to increase the paperwork but eliminates the need to revisit a global decommissioning plan in the case of change.

As evidenced by the participants’ desire to continue to participate and fund the Programme and the continuous growth, the Programme has proven very popular and useful. The United States DOE and the European Commission’s Joint Research Centre at Ispra have recently rejoined and three new members have been accepted into the Programme in the past five years; namely - Studsvik Nuclear AB (Sweden), Barsebäck Kraft AB (Sweden), and the Chubu Electric Power Company (Japan). The information acquired from the participating projects has influenced the aforementioned trends that are developing and has been a key factor in the selection of decommissioning techniques used by the projects within the programme. General examples of this include characterisation instrumentation and equipment, decontamination techniques, verification techniques, radiation protection, dismantling equipment and methodology and waste handling and storage

To address more general issues of common interest the CPD Technical Advisory Group established specific Task Groups. The Task Group on Decommissioning Costs, the Task Group on Recycling and Re-use, the Task Group on Decontamination, and the Task Group on Release Measurements have all completed their missions.

Two new Task Groups have recently been established. The Task Group on Decontamination and Dismantling of Concrete structures is examining all aspects of dealing with the immense volumes of concrete arising from the dismantling of nuclear facilities and fuel facilities in particular. The Task Group on Remote Handling Techniques is examining technological developments in this field and there application to dismantling of structures and equipment. The report from the Concrete Structures group is expected in late 2010 while the report from the Remote Handling group has suffered some setbacks but should follow in a few months.

Since its creation by the OECD Nuclear Energy Agency, the CPD has functioned as the leading international forum for the exchange of technical and other information arising from nuclear decommissioning projects. In addition to the tangible benefits listed above, personal interaction among experienced people from a wide cross-section of the decommissioning community is also a valuable benefit in itself.

As for the future, it is anticipated that the Co-operative Programme will continue to be well supported and that new projects, that will benefit the exchange of information, will be admitted to the Programme.

1. Introduction

The Co-operative Programme for the Exchange of Scientific and Technical Information Concerning Nuclear Installation Decommissioning has recently completed 25 years of operation. The Programme, which is also known under the short title Co-operative Programme on Decommissioning (CPD), was established in 1985 as a joint undertaking according to Article 5 of the Statute of the OECD Nuclear Energy Agency (NEA). Based on a specific Agreement between organisations actively executing, planning or having plans regarding decommissioning of nuclear facilities, the CPD has grown from an initial 10 decommissioning projects from 9 participating organisations to 59 decommissioning projects from 24 organisations today.

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

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

The initial Agreement inaugurating the Co-operative Programme on Decommissioning had been concluded for a five years term and was renewed in the years 1990, 1995 and 2000. However, in the year 2003 the agreement was updated to include a financial mechanism to support the work of a Programme-Co-ordinator and a new Agreement was concluded to come into effect in 2004, again for a five years period until the end of 2008. The current agreement, effective January 2001, 2009, will remain in effect until the end of 2013. The agreement in principle is unchanged from 2004 with the exception of a minor wording change that would specifically allow the European Union to join the programme as a participating member.

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

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

The Programme has published three overview reports earlier [2,3,4]. This report is intended to supplement the previous report which in fact covered a ten year period and was entitled "A Decade of Progress". That report covered the evolution of the CPD Programme over the first twenty years including the changes to the Programme introduced in 2003. This report will focus on the period 2005 to the end of 2010 and will include the activities of the Management Board, the Technical Advisory Group and the active Task Groups for that period. In addition, this report will highlight the significant lessons learned from the various projects as well as significant technological developments that promote efficient project execution.

The report includes a summary description of the participating projects in the CPD programme. The descriptions vary in length and detail, with significantly more detail being generally provided for those projects at an advanced stage or which are complete.

2. Structure of the co-operative programme

This chapter gives an overview of the structure of the CPD under the current Agreement (2009) and of how the various functions of the Programme and within the Programme have gradually evolved over the years.

2.1 Participating organisations

The NEA Co-operative Programme on Decommissioning functions within the framework of an agreement between a number of organisations actively executing, planning or with future plans regarding decommissioning of nuclear facilities. Currently, 24 organisations from Belgium, Canada, Chinese Taipei, France, Germany, Italy, Japan, Korea, Slovak Republic, Spain, Sweden, United Kingdom and the United States are participants of the Agreement of the Co-operative Programme on Decommissioning. In 2010, the agreement was slightly revised to enable the European Commission Joint Research Council (JRC) to join the Programme. The United States withdrew from the Programme in 2002 and has rejoined in 2010. Estonia, a former member no longer participates in the programme. Due to changes in some projects and mergers of organisations the number and identity of participating organisations has changed over time. A full list of current participants in the Co-operative Programme on Decommissioning is given in Table 1.

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

2.2 Management board

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

2.3 Technical advisory group

The central forum for the exchange of information is the Technical Advisory Group (TAG), which meets twice a year, generally at the site of a participating project. It is composed of technical managers and other senior specialists from the currently 59 member projects in the Programme. The projects are divided into two general groupings consisting of 35 reactor projects and 24 fuel-related projects. Out of the 59 participating projects, 43 of these projects are considered active (IAEA stage 1), 5 projects are considered in a dormant state (IAEA stage 2) and 11 of the projects are complete (IAEA stage 3). The TAG exchange forums generally focus on the active projects. (A full list of the projects currently participating in the Co-operative Programme is provided in Chapter 3 and more detailed information on the projects is given in Appendix 1.)

Table 1. Organisations participating in the Programme

Country	Organisation
Belgium	• Belgoprocess NV • Centre d'étude de l'énergie nucléaire/Studiecentrum vor Kernergie (CEN•SCK)
Canada	• Atomic Energy of Canada Limited/ Énergie atomique du Canada limitée (AECL/EACL)
Chinese Taipei	• Institute of Nuclear Energy Research (INER)
France	• AREVA NC[2] • Commissariat à l'énergie atomique (CEA) • Électricité de France (EDF)
Germany	• Energiewerke Nord GmbH (EWN)
Italy	• Società Gestione Impianti Nucleari SpA (SOGIN)
Japan	• Japan Atomic Energy Agency[3] (JAEA) • Japan Atomic Power Co. (JAPCO) • The Radioactive Waste Management and Nuclear Facility Decommissioning Technology Center (RANDEC) • The Chubu Electric Power Company
Republic of Korea	• Korea Atomic Energy Research Institute (KAERI)
Slovak Republic	• Jadrova a vvrad'ovacia spolocnost, a.a. (JAVYS)
Spain	• Centro de Investigaciones Energéticas, Medioambientales Tecnológicas (CIEMAT) • Empresa Nacional de Residuos Radioactivos SA (ENRESA)
Sweden	• Svensk Kärnbränslehantering AB (SKB) • Studsvik Nuclear AB • Barseback Kraft AB
United Kingdom	• Sellafield Ltd. • UKAEA Ltd
United States	• Department of Energy – Office of Environmental Management
European Union	• European Commission – Joint Research Centre Ispra

2.4 TAG Meetings

TAG meetings are held twice annually. The frequency of the meetings was chosen to ensure that the projects have ample opportunity to be updated at least once annually. Host organisations generally volunteer to host a meeting however if there is some particularly interesting work going on at a site, that organisation may be prompted to host the meeting. The meetings generally consist of three days of project progress reports and status reports from other activities

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

within the Programme. This is followed by one or two days of visits to the project sites for a first-hand look at the work in progress, tools, equipment and waste facilities. Site tours have proven very informative and provide an excellent opportunity for impromptu discussions on methods and techniques.

A summary of meeting sites during the term of the last agreement (2004 – 2008) and the first two years of the current agreement (2009 – 2013) and host organisations is provided in Table 2.

Table 2. Host and location of TAG meetings since 2004

Meeting No./Year	Location/Host Organisation	CPD Projects Featured
Tag 36 – 2004	Korea/Kaeri	Triga Mark II&III, Uranium Conversion Plant
TAG37 – 2004	Germany/EWN (Julich)	AVR
TAG 38 – 2005	Japan/JAEA	Fugen, Tokai 1, JRTF,PFPF
TAG 39 – 2005	France/CEA	ATUE Facilities
TAG 40 – 2006	Germany/Forschungszentrum Karlsruhe	MZFR, KNK,WAK, VEK
TAG 41 – 2006	UK/UKAEA	WAGR/ B204/B243
TAG 42 – 2007	Spain/Enresa	Pimic
TAG 43 – 2007	Germany/EWN	Greifswald/Rheinsberg
TAG 44 – 2008	Belgium/SCK/CEN/ Belgoprocess	BR3, Eurochemic plant
TAG 45 – 2008	UK/Sellafield Ltd.	B204, B243 (waste recovery), WAGR
TAG 46 – 2009	Korea/Kaeri	Triga Mark II&III, Uranium Conversion Plant
TAG 47 – 2009	Canada/Atomic Energy of Canada Ltd (AECL)	Whiteshell Research Labs (WRL) Decommissioning site.
TAG 48 – 2010	Sweden/Studsvik Nuclear AB and Barseback Kraft AB	Studsvik Nuclear Site Facilities.
TAG 49 – 2010	France/CEA	Marcoule Site - UP1 and Super Phenix.

The meetings are generally well attended and on the average 65% of Category 1 Projects (active projects) are represented at each meeting. The number of projects represented at each meeting in the past five years has increased (relative to the previous periods) due to the growth of projects in the Programme. There are now on the average 25 progress reports given at the meetings as opposed to approximately 20 reports in the previous period.

The actual number of attendees has also slightly increased on the average. Attendance can be influenced by the meeting location. Meetings in central Europe are generally better attended as travel to these locations is less expensive for the majority of the members. Timing of the meeting (around other conferences, workshops etc) can also affect attendance. The fall is always a busy time for other events and although the meeting dates are chosen to avoid conflicts, it is not always possible. Average attendance at the meetings over the past 5 years is just shy of 30 people. The fact that the attendance has not increased proportionally to the increase in the number of projects can be attributed to tighter control on attendance – unsolicited observer status attendance has been significantly reduced.

Due to the increase in the number of project reports, it has recently been necessary to introduce tighter control of the meetings in order to make the most efficient use of the allocated meeting time. The following measures have been introduced:

- Each project has to name a single Project Representative who is the single point of contact for CPD/TAG information. It is the responsibility of the Project Representative to further disseminate information within the project group as necessary. The Project Representative list is maintained by the Project Co-ordinator
- Participants are strongly urged to limit historical information to a couple of slides and focus on progress since the last report – if there has been no progress or new developments a statement to this effect is all that is necessary. There are no time limits on presentations in order not to suppress the exchange of new information.
- With the exception of the host organisation, members are urged to send only one representative for each project to the meeting. A large number of non participating observers can be disruptive and puts a heavier burden on the host organisation
- Participants are required to record their presentations on a memory device. All presentations are then loaded onto a master computer prior to the meeting for the presentations. Power Point Presentation is the preferred media.
- With the help of the host organisation, multiple copies of the presentations are made (CD, DVD or USB key) and distributed at the meeting for the participants to take with them. This has eliminated the cumbersome stacks of hard copy material.
- Host organisations are urged to organise meeting locations so that travel time for site visits can be minimized.

The aforementioned control measures have in fact resulted in enough time to hold an informal topical session at the TAG meetings. These topical sessions focus on general issues common to several of the participating projects and provide an opportunity for the participants to present their experience relative to the topic in detail. Examples of topical sessions over the past couple years are:

- Radiological Characterisation.
- Current Practices for Soil Characterisation and Remediation.
- Development of new/innovative characterisation techniques.
- Ground Plume Monitoring – Chemical and Radiological.

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 for review 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. After acceptance by the TAG participants, the summary records are posted on the CPD protected website.

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. The TAG meetings provide the forum for exchange of information and facilitate broader and more detailed collaboration and follow-up between participants.

2.5 Task groups

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

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

Task groups have worked in the following areas:

- Decommissioning costs [6].
- Recycling and re-use of slightly contaminated material from decommissioning [7].
- Decontamination in connection with decommissioning [8].
- Release measurements [9].
- Remote handling devices (2008) [10].
- Decontamination and dismantling of concrete structures (2008).

The report from the Task Group on D&D of concrete structures will be published late in 2010 and the report from the Task Group on remote handling devices is expected to be published in 2011.

2.6 Programme co-ordinator

From the start of the Co-operative Programme, the smooth functioning of its arrangements and procedures had been ensured by the appointment of a Programme Co-ordinator. 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. Prior to 2001 these services were provided by Sweden (SKB). However, SKB at that time decided to reduce its support to the Co-operative Programme and hence also the provision of the Programme Co-ordinator. The participants of both the TAG and the Liaison Committee concluded that for the continued efficient running of the Co-operative Programme, there was a need for the continued role undertaken by the Programme Co-ordinator. It was decided to ask each participant in the Programme to contribute to funding for the provision of a Programme Co-ordinator. This was duly agreed by the LC participants and so, since 2001, a financial contribution from each member in the CPD is sought to cover the costs of programme co-ordination.

2.7 Link to the NEA committee structure

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

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

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

It is important that there is a high degree of co-operation and cross fertilisation between the WPDD and the CPD. To ensure this close relationship some delegates are members of both committees and bureaux, and co-operation between both groups has been formalised by an interface document [5].

3. Programme activities

3.1 Projects participating in the information exchange

In the past five years the CPD programme has experienced significant growth both in terms of the number of participants and a corresponding increase in the number of projects. The United States rejoined the programme in 2009, bringing five previous projects back and also introduced a new project. Barseback Kraft and Studsvik Nuclear (Sweden), the Chubu Electric Power Company (Japan) and the European Union also joined the programme each bringing new projects with them. In addition, nine new projects were introduced by existing members. The Prototype Fast Reactor (PFR) project at Dounreay (UK) opted out of the programme. The CPD project growth over the past five years is summarized below.

New Reactor Projects:

1. Whiteshell Research Laboratory, AECL, Canada (2008).
2. Phenix, CEA, France (2007).
3. Hamaoka Units 1, 2, Chubu Electric Power Company, Japan (2010).
4. Bohunice V1 Npp's, Slovenske Elektrarne, Slovakia (2007).
5. Jose Cabrera Npp's, Enresa, Spain (2010).
6. Pimic D&D, Enresa, Spain
7. Barseback Npp's, Barseback Kraft, Sweden (2007).
8. Studsvik Research Reactor, Studsvik Nuclear, Sweden (2007).

New Fuel/Fuel Related Projects:

1. ISPRA Legacy Waste Recovery, EU/JRC, Italy (2010).
2. Saclay NLF, CEA, France (2009).
3. Pilot U-Th Reprocessing Plant, SOGIN, Italy (2008).
4. Uranium Refining/Conversion Facility, JAEA, Japan (2008).
5. B 243 Intermediate Waste Recovery, Sellafield Ltd, UK (2007).
6. Portsmouth Gaseous Diffusion Plant, DOE, USA (2009).

Projects Re-instated:

1. Shippingport, DOE, USA (2009).
2. EBWR, DOE, USA (2009).
3. Fort St.-Vrain, DOE, USA (2009).
4. West Valley, DOE, USA (2009).
5. FEMP, DOE, USA (2009).

Projects Opting Out of the Programme:

1. Prototype Fast Reactor, UKAEA, UK (2009).

As of December 2010, 59 individual decommissioning projects are participating in the information exchange of the Co-operative Programme on Decommissioning, up from 42 in 2006. They cover 35 reactor decommissioning projects and 24 projects on decommissioning of fuel cycle and other fuel related facilities.

The projects in the Programme have a broad range of characteristics and cover various types of reactors and fuel facilities. A full list of the projects that are or have been participating in the information exchange of the CPD is given in Table 3 (reactor projects) and Table 4 (fuel cycle facility projects). For these projects, some general comments can be given:

- The reactors represent a wide selection of types such as PWR, BWR, PHWR, gas cooled/D2O moderated, water cooled/D2O moderated, GCR, AGR, VVER, sodium cooled fast reactors and HTGR's both with block type and pebble bed fuel design. The list of reactor projects formerly also included the decommissioning of a plant with two Russian submarine reactors.
- Of the 35 reactor projects, 6 have been completed, i.e. decommissioned to Stage 3, and 5 have been placed in a "dormancy" status (Stage 2 or Stage 1). Of the 24 fuel related projects, 5 have been completed. Stage 3 implies that the sites have been returned to "green field conditions" or decontaminated completely so as to have been removed from regulatory control. The completed projects continue to be considered as being part of the Programme, as the information arising and the experience from these projects are in the Programme archives. The dormant plants can also continue to generate information and experiences on building/plant degradation and long-term surveillance.
- Many of the earlier projects in the Programme had to do with experimental or prototype plants. The projects, which have joined the Programme at a later date, were, for understandable reasons, related to plants of a more standardised and commercial character. Even so, there are still significant differences that can be seen in the planning and execution of decommissioning projects. Apart from the differences that can be expected due to the variation in type of plant, the organisational, economic, regulatory and other circumstances prevailing at each site can strongly influence the decommissioning projects.

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

Table 3. Reactor Projects Participating in the CPD

	Facility	Type	Operation	Stage (IAEA)	Power	Timescale	Cost Estimate	CPD Entry	Remarks
Category 1 - Active Projects									
1	BR-3, Belgium	PWR	1962-1987	3	41MWt	1989-2020	150M€ (2000)	1988	EC pilot project
2	Whiteshell Res. Lab, AECL Canada	Organic cooled, HW moderated	1965-1985	1,2,3	60MWt	2002-2060		2007	Only site in Canada with full decomm. licence
3	EL4, Brennilis, France	Gas cooled, HW moderated	1966-1985	2	70MWe	1989-2017		1993	
4	Bugey 1, France	Gas graphite reactor	1972-1994	3	540MWe	1997-2021		2004	
5	Melusine, France	Pond Research Reactor	1988-1993	3	8MWt	1999-2009	20M€ (2003)	2004	
6	Phenix, France	FBR sodium cooled	1973-2009	3	250MWe	2009-2023	650M€ to 850M€ (2007)	2007	Without civil works demolition.
7	MZFR, Germany	PWR, HW cooled and moderated	1965-1984	3	57MWe	1984-2014	330M€ (2009)	1989	Waste costs N/I
8	Greifswald	VVER (eight units)	1973-1990	3	440MWe (five units)	1990-2013	1653M€	1989	Waste storage, disposal costs N/I
9	AVR Germany	Pebble bed HTGR	1967-1988	3	15MWe	1994-2015		1994	
10	KNK, Germany	Fast breeder reactor	1971-1991	3	20MWe	1991-2013	310M€ (2007)	1997	Waste costs N/I
11	Garigliano, Italy	BWR (dual cycle)	1964-1978	3	160MWe	1978-2020	297M€ (2000)	1985	Dormancy in 1999
12	Latina, Italy	GCR (Magnox)	1963-1986	3	210/160MW	1986-2020	615M€ (2000)	1999	
13	Fugen, Japan	Light water cooled, HW moderated	1979-2003	3	165MWe	2008-2028	74.7M JPY (2008)	2000	Advanced thermal reactor
14	Tokai, Japan	GCR	1966-1998	3	166MWe	2001-2020	88.5M JPY	1997	Waste costs N/I
15	KRR-1 & 2, Korea	Pool type research reactors (Triga 1 & 2)	1962-1995 1972-1995	Stage 3	250 KWt 2 MWt	1997-2013	23.2 MUSD	1997	Without costs for radwaste disposal
16	Hamaoka, Japan, Unit1 Unit2	NPP - BWR	1976-2009 1978-2009	3	540MWe 840MWe	2009-2036	379M USD 462M USD	2010	To be replaced by Unit 6
17	Bohunice A1, Slovakia	Gas cooled, HW moderated	1972-1979	3	150MWe			1992	S/D Fuel accident
18	Bohunice V1, Slovakia	PWR Unit 1 Unit 2	1976-2008 1978-2008	3	408MWe 408MWe	2012-2025	470M€	2007	
19	Pimic, Spain	MTR Reactor	1958-1984	3	3MWt	1999-2008	22.5M€	2006	Includes other CIEMAT facilities
20	Jose Cabrera NPP, Spain								
21	Studsvik, Sweden, R2 R2-0	Light water, MTR tank-type pool reactor	1960-2005	3	50MWt 1MWt	2007-2024	57.3M€	2007	Now in care and maintenance

Table 3. Reactor Projects Participating in the CPD

	Facility	Type	Operation	Stage (IAEA)	Power	Timescale	Cost estimate	CPD entry	Remarks
Category 1 - Active projects									
22	Barseback, Sweden Unit1 Unit2	BWR	 1975-1999 1977-2005	3	 615MWe 615MWe	2007-2024	167M€	2007	Now in care and maintenance
23	TRR, Taiwan	Light water cooled, HW moderated	1973-1988	2	20MWt	1998-2028	9000M NT$	2000	
24	WAGR, Sellalfield, UK	AGR	1962-1981	3	100MWt	1983-2015	80M GBP	1985	EC pilot project
Category 2 – Stage 2 (dormancy)									
25	Gentilly 1, Canada	HW moderated, light water cooled, prototype	1972-1978	2	250MWe	1984-1986	25M CAD (1986)	1985	In dormancy
26	NPD, Canada	PHWR CANDU Protoype	1962-1987	2	25MWe	1987-1988	25.3M CAD	1988	In dormancy
27	Rapsodie, France	Experimental sodium cooled, fast breeder reactor	1967-1982	3	40MWt	1983-2020	127M€ (2005)	1985	
28	G2/G3, Marcoule, France	GCR, NPP and nuclear materials production	1958-1980	2	250MWt (each)	1982-1983	150M FRF (1990)	1985	Stage 2 achieved
29	Vandellos 1, Spain	GCR	1972-1989	2	460MWe	1998-2003	96M€	1993	In dormancy
Category 3 – Stage 3 (complete)									
30	KKN, Germany	Gas cooled, HW moderated	1972-1974	3	0.6MWe	1988-1994	134M€	1985	Waste cost N/I
31	HDR, Germany	BWR, nuclear superheat	1969-1971	3	100MWt	1993-1998	150M€	1993	Waste cost N/I
32	JPDR, Japan	BWR	1963-1976	3	90MWt	1986-1996	22M JPY	1985	Stage 3 achieved
33	Shippingport, USA	PWR (2 cores) LWBR (1 core)	1957-82	Stage 3	72 MWe	1984-1989	$98 M (US)	1985	Stage 3 achieved Actual cost $72.5 M
34	Exp. Boiling Water Reactor (EBWR), USA	BWR	1956-67	Stage 3	100 MWt	1986-1996	$19.5 M (US)	1990	Stage 3 achieved.
35	Fort St. Vrain, USA	High Temperature Gas Cooled	1976-1989	Stage 3	170 MWe	1989-1996	Not Available	1993	Stage 3 achieved.
Former reactor projects not now participating in the programme									
	Lingen, Germany	BWR	1969-1971	1	520MWt	1985-1988		1985	
	Paldiski, Estonia	Soviet Submarine PWR	1966-1985	1		1994-?		1997	
	PFR, Dounreay, UK	Sodium Cooled, fast breeder	1974-1994	1	250MWe			1997	

Table 4. Fuel and Fuel Related Facilities Projects Participating in the CPD

	Facility	Type	Operation	Stage (IAEA)	Power or throughput	Timescale	Cost Estimate	CPD Entry	Remarks
Category 1 - Active Projects									
1	Eurochemic Reprocessing Plant, Belgium	Reprocessing of fuel	1966-1974	3	300 Kg/d	1989-2013	175MEUR (2008)	1988	Execution by in-house staff
2	Building 204, Bays, Canada	Fuel storage pond	1947-1996	2				1997	
3	ISPRA legacy Waste Recovery, EU, Italy	Research Facility	1959-	3		2010-2028	~32M€	2010	Disposal costs N/I
4	Radio Chemistry Laboratory, France	Reprocessing R & D	1961-1995	3		1995-2015	262M€	1999	Nuclear Centre inside City
5	ATUE France	Recovery of enriched uranium	1965-1996	3		2000-2012	5 M€	2004	
6	Elan IIB France	Manufacture of ^{137}Cs & ^{90}Sr sources	1970-1973	3		1981-2018	90M€	2000	
7	APM, Marcoule France	Pilot reprocessing plant	1965-1997	3	5t/annum	1997-2020	840 M€	2004	
8	UP1, Marcoule, France	Industrial reprocessing plant	1958-1997	3	500 t U/a	1998-2032	2600 M€	2002	Main project in France
9	Saclay NLF, France	R&D U enrichment, isotopes	1957-1996	3		2000-2018	2800 M€	2009	
10	WAK, Germany	Prototype reprocessing plant	1971-1990	3		1991-2023	2170 M€ (2008)	1993	Waste disposal N/I
11	SOGIN – PilotU-Th Reprocessing Plant	Reprocessing and fabrication of fuel	1975-1978	3		2003-2018		2008	Cost and timetable under revision
12	JRTF, Tokai Japan	Reprocessing test facility	1968-1970	3		1991-2026	8600M JPY	1991	
13	Plutonium Fuel fabrication Facility, (PFFF) Japan	R & D on plutonium and MOX fuels	1972-2002	3		2005-?		2004	
14	Uranium Refining, Conversion, Enrichment Facilities	Uranium refining /Conversion Uranium enrichment	1981-1999 1986-2001	2	120tU/y 200tSWU/y	2008-2016	10 M€ 15 M€	2008	First case (commercial scale) in Japan.
15	Uranium Conversion Facility, Korea	Conversion of yellowcake to UO2//UF4	1982-1992	3	100 t U/a	2001-2010	9.6M USD	2003	Waste cost N/I

Table 4. Fuel and Fuel Related Facilities Projects Participating in the CPD (continued)

	Facility	Type	Operation	Stage (IAEA)	Power or throughput	Timescale	Cost Estimate	CPD Entry	Remarks
Category 1 - Active Projects									
16	BNFL, B204 Primary Separation Plant, UK	Reprocessing facility	1952-1973	2	Metal-500t/a Oxide-140t/a	1990-2090	MGBP 570 (2010)	1990	Cost excludes Stack Demolition and new Stack construction
17	Sellfield – B243 Intermediate Waste Recovery	Solid Waste Storage Cells	1970-1986	2	N/A	2000-2040		2008	
18	Portsmouth Gaseous Diffusion Plant, USA	High and Low Enriched Uranium	1954	1	300 M Kg Uranium	2010-2044	\$5 B - \$12 B	2009	
19	West Valley, Demonstration Project, USA	Reprocessing facility for LWR fuel	1966-72	Stage 1	100 t/a			1986	Facility was dormant for several years, re-starting in 2010
Category 2 - Stage 2 (dormancy)									
	None								
Category 3 - Stage 3 (complete)									
20	Tunney's Pasture Facility, Canada	Isotope handling facility	1952-83	Stage 3		1990-94	MCAD 13 (1991)	1990	Stage 3 achieved
21	BNFL Co-precipitation Plant, 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
22	AT-1, La Hague, France	Pilot reprocessing plant for FBR	1969-1979	Stage 3	2 Kg/d	1982-2001	80 M€ (2007)	1985	EC Pilot Project
23	AB SVAFO ACL Project, Sweden	PU & enriched fuel research	1963-97	Stage 3		1998-2006	8M€	1999	Stage 3 achieved
24	Femald Environmental Mgmnt Project, USA	High Purity, low enrichment uranium reactor feed material	1952-1989	3		1989-2006	2.3B USD	1993	Stage 3 achieved
Former Fuel Facility Projects not now participating in the Programme									
	None								

3.2 Decommissioning projects

One of the key characteristics of decommissioning projects is that no two projects are identical. Although some projects in the same grouping (reactor types for example) may have many commonalities, the whole life cycle of the facility including construction, operation, modification, maintenance and record keeping will ultimately determine the associated hazards and complexity of the characterisation and dismantling processes. Likewise, national and local nuclear waste policies including waste handling/transporting and the availability of storage or disposal facilities can largely influence schedule, cost and strategic planning. Decommissioning nuclear facilities has been going on for several years; several large projects have been completed or are at a very advanced stage. The experience gained from these projects continues to grow and as this information is disseminated through programmes such as the CPD, the efficiency of project planning and execution continues to improve, benefitting cost, schedule and safety.

Looking back over the past 25 years, some general conclusions can be made:

- Decommissioning can and has been done in a safe, cost-effective and environmentally friendly manner with a progressive improvement being demonstrated with experience.
- Current technologies are adequate but a continued R&D programme is necessary for continued process imprvement.
- Feedback of experience on design, construction and operation is key for reliable planning, cost evaluation and successful realisation of a decommissioning project.
- The CPD and similar programmes have effectively improved decommissioning processes and efficiency. With decommissioning moving towards being a fully mature industrial process, there is a need for increased dialogue and co-operation between the various programmes and amongst regulators, implementers and international standards organisations.

Perhaps the most dominating factor influencing the establishment of general national decommissioning strategies is the vast quantities, and varieties, of waste that are generated from Decontamination and Demolition (D&D) of nuclear facilities. In the early years of the CPD Programme, many countries anticipated that national disposal facilities would be available in the next couple of decades and a common approach was to bring a shutdown facility to a safe state (Stage2) awaiting the availability of waste disposal facilities and also taking advantage of the radioactive decay (source reduction) in the meantime. The establishment of waste disposal facilities has not materialized as fast as anticipated (with a few exceptions) and in many cases are still decades away. With the acceptance that the deferral approach was proving not to be cost effective nor was it a popular solution from the public perspective, more and more projects are dealing with the waste issues either through intensive waste reduction programmes or developing interim solutions to deal with the waste, allowing the project to advance efficiently to completion.

These developments have influenced all facets of both ongoing projects and upcoming projects including basic strategies at both the national and project levels. The majority of new projects are fast-tracking the D&D scenario and many of the older projects are changing strategies mid-stream to do the same. Many of the older research facilities and prototype reactors are already in the decommissioning stream however a large number of power reactors will be reaching the end of their lifetime in the next couple decades. The number of active decommissioning projects should grow significantly over the next couple decades, as will the return of experience (REX). The CPD has some exciting years ahead!

3.3 Activities of task groups

Two task groups have been established within the CPD during this reporting timeframe. Namely: the Task Group on D&D of Concrete Structures and the Task Group on Remote Handling Techniques.

3.3.1 Task group on D&D of concrete structures

This task group was established in 2008 in recognition of the fact that concrete rubble from the demolition of buildings and other structures was a major contributor to decommissioning waste streams and that the characterisation and cleanup of this material presented significant challenges. The task group report was initially scheduled to be published at the end of 2009 but due to the complexity of this subject and the volume of material collected, the report is now published at the end of 2010 [5].

3.3.2 Task group on remote handling techniques

This task group was also established in 2008 to examine and report on the application of remote handling devices to dismantling tasks. Unfortunately this task group suffered some early setbacks due the necessary withdrawal of several key volunteers on the team. A significant volume of information has been collected and this information is currently being assessed to see if it is worth while continuing with this task group. Early assessments are positive and it is likely that the report will be published in 2011.

4. Benefits of the CPD programme

The objective of the CPD is to acquire information and share operational experience from the conduct of the various decommissioning projects. As mentioned previously decommissioning projects may share some comment aspects in terms of general hazards or waste issues however each facility has its own unique history of construction, operation, accidents and repair. Thus one cannot expect to duplicate a project, however over the years basic strategies, sequence of events, and methodologies have evolved and various pitfalls have been identified that can be avoided. As a result of the REX from different projects, it is apparent that there are now many commonalities in the execution of the various projects. Feedback from the projects has provided the basis for much more accurate input into ever improving project planning systems and databases resulting in performance improvements in terms of cost, schedule, safety and environmental impacts.

Some specific trends observed and previously reported [3] such as large component dismantling, sequential licensing as opposed to a global licence, and the use of robotics for niche applications as opposed to general use are continuing. More recently there has been a strong focus on reduction of radioactive waste through intensive programmes of characterisation, decontamination and release of both metal and concrete components as conventional waste or for recycle/reuse. This strategy has proven cost effective as disposal costs continue to escalate, particularly in situations where the availability of disposal facilities is many years away. Many projects are including within their scope the design and construction of engineered combination interim storage facilities and waste processing facilities where the waste is further reduced through cutting of larger components for decontamination and free-release. These trends are discussed in more detail below.

4.1 REX/Lessons learned and influence

4.1.1 Prompt decommissioning vs. deferral

Deferred decommissioning was once a popular choice due to lack of waste disposal facilities, lack of experience and expertise (buy some time) and the benefits of radioactive decay (source reduction). Over the years experience has proven:

- The cost of maintaining a nuclear facility in a safe state (Stage2) is very costly. Regulators demand a stringent safety regime in regards to radiation monitoring and protection, fire prevention, industrial safety, seismic qualification, materials management and security.
- Costly heating ventilation systems may have to be installed and maintained.
- Building structures deteriorate and must be maintained (e.g. roofing). For example, B204 at Sellafield requires a new emergency exit staircase, incurring a high cost.
- Equipment and process systems corrode and decay, often leaking contaminants into the building structure that must be later cleaned up at high cost.
- Many old process systems used asbestos insulation. The containment eventually deteriorates and asbestos may spread throughout the building, resulting in significant additional cost for cleanup (Sogin/Garigliano for example).
- Waste disposal and storage costs are rapidly escalating with time.

- Experienced and knowledgeable operating staff disappear and poor historical records deteriorate even further.

Modern technology, improved work processes and emphasis on safety have negated the advantages of long term deferral of decommissioning. Likewise, improved dismantling techniques including remote operation, recycle/reuse, development of efficient interim waste storage and processing facilities, and improved planning processes have made prompt decommissioning more attractive. The prompt decommissioning option is further enhanced by starting the decommissioning planning process several years before operational shutdown Phenix for example). This is generally the preferred option from a public perspective.

The large EWN reactor decommissioning project at Greifswald has demonstrated that prompt decommissioning resulted in lower overall cost, lower dose commitment, lower radioactive waste production and has minimized the social impact by re-employing a significant percentage of the operations staff.

The majority of new projects are chosing the prompt decommissioning option. As well, several projects that had originally chosen the deferred option are reversing this decision (AVR, Fugen, Brennilis, Melusine, Garigliano - to mention a few).

4.1.2 Decommissioning planning

Planning is the cornerstone of a project and can make or break the project. The plannning process for early projects was hampered by several factors:

- General lack of decommissioning experience, both on the project management side and the regulatory side.
- Lack of firm national policies or strategic direction.
- Corporate restructuring and reorganisation as the decommissioning era began.
- Little interaction between operational regime and decommissioners.
- Design, construction and operation of early facilities had little consideration for decommissioning issues.
- Lack of complete design and construction records and incomplete recording of operational history
- Loss of operational staff and experience.

Given these conditions, most projects have little likelihood of being able to plan and forecast the entire project and will face several iterations of planning, significantly impacting budgets, schedules and end state. Fortunately, after two or three decades of decommissioning nuclear faculties, significant progress has been realized. Although there are still some uncertainties at the national levels in terms of general nuclear policies and decommissioning strategies, the general trend toward prompt decommissioning has given some stability to the planning processes in that the projects are generally being planned over a shorter term rather than have a very long period of dormancy which is very difficult to plan through.

After over three decades of decommissioning a significant experience base has developed for input to the planning processes. The key issues and lessons learned from this experience include:

- Start the planning process early, prior to the shutdown of the facility if possible (Phenix for example). Involve the operational staff in the planning process and use the experience of veteran operational staff.
- Involve the Public early and use their input. Projects such as Brennilis have suffered serious delays from lack of public involvement.
- Involve the authorities early in the planning process – consider them an ally, not a deterrant.

- There is good cooperation between international organisations with similar objectives (IAEA, European Commission for example). Use the information available from all sources.
- As databases continue to be populated, the use of modern planning systems, both commercially available and those developed in house, are becoming more efficient and accurate with input from the databases. Projects such as BR3, MZFR, and Fugen are examples of projects that have made good use of planning packages. Publications such as the Proposed Standardised List of Items for Costing Purposes (The Yellow Book) (4) have proven useful.
- Depending on individual situations, a sequential licensing scheme may have advantages over a global licence structure. Under a sequential scheme, a global licence revision is not necessary if/as unexpected developments occur.
- A good materials management system is necessary to deal with large material volumes, prevent bottlenecks and delays and to optimize the material evacuation routes.
- An extensive and thorough characterisation of the facilities is crucial. Modern equipment including the use of remotely controlled cameras and laser imaging systems are greatly facilitating the characterisation process. Use historical records and veteran operational staff (including retirees) as input to the characterisation process.
- Experience is only useful if it is shared. The continued cooperation between international organisations and the use and growth of programmes such as the NEA/CPD is essential to ensure a continued decommissioning improvement process.

4.1.3 Interim waste storage facilities

With the uncertainty around the availability of radioactive waste disposal facilities, many projects are including the construction of interim waste storage and processing facilities within the project scope. Engineered waste storage facilities offer several advantages:

- Properly engineered facilities will safely contain the source. Acceptance criteria ensure proper packaging and inventory control. Building design and waste management procedures will minimize decommissioning efforts in the future.
- Interim Storage facilities can also be qualified for fuel storage (Greifswald for example).
- Waste stream bottlenecks can be eliminated or minimized.
- Large components can be removed from the D&D site sooner reducing the source term and improving the work conditions (more space).
- Declassification and release of the D&D site, including grounds, can be achieved much faster if the components are removed for further processing.
- If the storage facility includes waste processing facilities (hot/cold cutting, decontamination), cutting and decontamination campaigns can be carried out in an orderly, efficient manner (waste reduction).

The effectivness and efficiency of Interim Storage Facilities have been well demonstrated by the German projects at Greifswald and MZFR at Karlsrue, and the Belgium BR3 project.

4.1.4 Large component removal

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

Against this background and the demonstration that interim storage and processing facilities are cost effective and efficient, more projects are choosing to remove large components “in one piece” and to segment and package them in waste containers in separate facilities outside the containment. This approach is also being extended to removing the entire reactor pressure vessel, and the internals, in one piece through the use of special shielding and transport devices for storage and further processing in the future (Greifswaldt, AVR, TRR for example.

4.1.5 Manual vs. remote (robotics) dismantling

As previously reported in detail (3) the use of remote handling devices or robotics certainly has a place in the decommissioning industry particularly in areas where there are high radiation fields or other elevated industrial hazards. However, this technique has proven not to be as globally effective as once anticipated. Large scale remote devices have successfully been used to dismantle reactors (WAGR, MZFR for example) or other components in high radiation fields. These devices tend to be one of a kind and generally not reusable. Remote device development now generally focuses on smaller applications such as the control of cameras, laser imaging devices and sampling tools for characterisation of confined spaces especially in areas of high contamination and radiation fields (fuel processing cells or liquid waste storage tanks for example). Attempts to apply automation to general tasks such as concrete surface cleanup, general dismantling and general decontamination proved not effective. The adaptation of industrial available tools and the control of these tools through remote hand held devices had improved efficiency but however physical presence and overall control by the operator where possible is still the most effective approach.

Some of the key factors leading to this conclusion are:

- Automated devices normally involve long, complicated manipulator arms that are prone to breakage, are high maintenance items and costly. Availability time may be low.
- Extensive training programmes may be required.
- Applications such as cleaning or removing material from concrete surfaces are hampered by obstructions and complicated geometry which does not lend itself to automation.
- Adaptation of standard industrial equipment, improvements to cutting materials, use of mechanical equipment as carrier devices (Brokk, Forklifts etc) and improvement to controls of these devices has improved the efficiency of manual execution.
- Improvements to ventilated clothing and general improved radiation protection practices has reduced dose burden and increased general worker comfort in manual operations.
- The trend toward large component removal discussed earlier and the reduction of radioactive source materials has reduced the demand for automated dismantling devices.

In summary remote controlled devices and robotics still play a very important role in the decommissioning industry but the focus is now more toward specific applications as opposed to a general global application.

4.1.6 Mock-up, testing

Some projects, especially those using large and complex remote dismantling systems, have reproduced scale models of mechanical structures to prove the feasibility of planned dismantling operations (WAGR, MZFR, WAK for example). Increasingly popular is the use of 3-D laser imaging and CAD systems to re-create a virtual image of a complex or concealed structure (JRTF for example). These modern systems can be used to simulate a total dismantling scenario including the evacuation of materials through various routes. Interference situations can be identified well in advance of the actual operation.

4.1.7 Waste reduction (recycle/reuse)

The general lack of waste disposal facilities, ever increasing waste storage and disposal costs and a favourable political and public attitude toward recycling or converting nuclear waste to conventional waste has resulted in a very keen interest in waste nuclear waste reduction. Large projects such as the Belgoprocess/Eurochemic project, the SCK•CEN BR3 project and the INER TRR Project (amongst others) have demonstrated that close to 90% of the D&D materials produced can be recycled or disposed of as conventional waste at saving up to 50% over dealing with this material as radioactive waste.

Waste reduction achievements have been driven and supported by:

- The aforementioned lack of disposal facilities and high cost of waste storage and disposal.
- Waste reduction considerations are now normally entered into the planning process and shutdown operations processes in the early stages. Decontamination and source reduction are key components of the decommissioning preparation process and should be factored into budgets and schedules.
- Decontamination techniques, equipment and cleaning materials are becoming more successful and efficient with experience and R&D developments further driving the cost advantage.
- The development of interim storage facilities with dedicated waste processing facilities has significantly facilitated the waste reduction process.
- The development of specialized decontamination facilities (Belgoprocess bead blasting and concrete drum cleaning facilities for example).
- The continuously evolving R&D programmes supporting the development of improved radiation monitoring equipment, improving monitoring processes and techniques and an ever growing experience base is giving authorities a higher confidence level in monitoring results, facilitating the overall release process
- With increasing confidence levels and generally positive public acceptance, it is likely that more countries will establish release criteria.

Several major projects have been completed or are approaching the final phase of the project that encompasses the demolition of the buildings that housed the nuclear facilities. Concrete rubble is an ever increasing contributor to the waste volumes that have to be dealt with. There has been no singular major breakthroughs in dealing with the massive quantities of concrete but a continuous general improvement to materials used in cutting devices and cutter configuration have improved efficiency. For example, new concrete shavers are efficient in removing surface layers of concrete while leaving a relatively smooth surface which is easier to monitor, new, harder cutter materials have also made dry cutting practical reducing secondary waste and modern equipment has significantly reduced the vibration hazards from prolonged use. Laser applications to surface cleaning are under development (CEA – Apilaser for example).

It has been recently demonstrated that up to 90% of contaminated concrete structures can be cleaned and free released. Again, experience has improved efficiency. For example, a common practice is now to cut out sections of concrete walls that contain pipe or other penetrations and deal with these relatively smaller pieces apart from the main building structure. The concrete is broken away from the penetrations and decontaminated as a small component. Only small quantities of concrete in direct contact with penetration need be treated as radioactive waste. Likewise, similar techniques have been developed to deal with cracks and other irregularities in concrete structures. The Belgoprocess/Eurochemic project has successfully developed a total process for decontaminating, monitoring, cutting, crushing, sampling and finally free releasing large quantities of concrete as have other projects.

The current Task Group Report on D&D of concrete structures covers all aspects of dealing with concrete structures including the free release of concrete waste. This report will be published in late 2010.

4.1.8 Project management

Management of decommissioning projects presented many challenges particularly at the onset due to the lack of expertise and experience. The concept of outsourcing the entire project scope or large portions of it was popular but proved to have its shortcomings:

- Due to lack of experience and detailed knowledge about the facility, contract specifications were loose and poorly defined leading to dispute, work stoppages and significant overrun in cost and schedule.
- Contractors were not familiar with regulatory requirements leading to licensing issues.
- Contractors generally did not make use of experienced operations staff often resulting in permanent loss of this experience.
- Contractors generally did not attempt to minimize local social impacts of plant closures.
- Management functions were often duplicated. For example, both the facility owner and the contractor might have radiation safety management structures.

Over time, again as the experience base built up, many of these problems have been overcome or minimized. The key to success is to plan your project thoroughly and accurately (as discussed above). A very detailed and precise contract specification is required with close scrutiny and control of the project execution.

Recently there has been a trend to perform the work in-house to the extent possible. The use of ex-operational staff and local small contractors for small packages of work has alleviated the social impacts and resulted in a generally more amicable atmosphere. The JAEA Uranium Refining/Conversion Facility project has realised cost savings of close to 50% in some areas with a switch from lump sum contracting to local company staffing and temporary small contractors.

There is some evidence that decommissioning programmes have been adversley affected by political issues, such as changes in national strategy and limitations to funding.

4.2 Conclusion

The benefits of the CPD Programme discussed above are of interest and may be applied at a general level. In addition to these benefits there are many project specific problems that are put on the table, discussed and often solved at TAG meetings and though personal interaction as a result of TAG meetings. Many of these specific issues are discussed in the individual project summaries appended to this report.

After 25 years of operation, some general conclusions can be drawn in regards to the CPD Programme specifically and to international co-operation in general:

- CPD has proven to be a good basis for an effective co-operation and support, to master new challenges in decommissioning projects.
- CPD has worked to avoid discrepancies, to save money and has helped promote reliable planning, cost evaluation, and safety.
- The working method of CPD/TAG encourages openness between members in sharing experiences and ideas.
- Increased dialogue among regulators, implementers and international standards organisations is necessary.
- The dissemination of best practices and sharing of information in international workshops, conferences and especially within the CPD has proven to be a good basis for

an effective cooperation and support to master new challenges in decommissioning projects;

- Continued international cooperation is important for meeting future challenges in decommissioning.

5. Future of the Co-operative Programme

The CPD has functioned as the main international forum for the exchange of technical and other information arising from nuclear decommissioning projects for the past twenty five years. Several projects within the CPD Programme have reached conclusion, releasing from regulatory control (Stage 3 decommissioning) a number of diverse nuclear facilities. These projects have benefitted from the programme and likewise have returned valuable information to the programme for the benefit of other members. Through the projects in the programme it has been demonstrated that decommissioning can be performed safely both for workers and the public, and in an environmentally friendly fashion. Through the benefits of the Programme, this is being accomplished with increasing efficiency.

The rise in participation in the Co-operative Programme on Decommissioning from 10 projects in 1985 to 59 projects 25 years later and this increasing participation reflects the success and popularity of the Programme. It is anticipated that the Programme will continue to attract new participants and that this will lead to a broader exchange of scientific and technical exchange as new challenges are presented and new solutions evolve.

Given the success of the Programme, there are no plans to change its basic structure and operation in the foreseeable future. TAG Meetings will remain the main forum for the presentation of new processes, techniques, strategies, tools and other pertinent information relevant to decontaminating and dismantling Nuclear Facilities. The Programme Co-ordinator and Project Representatives are responsible for facilitating follow-up collaboration. TAG meetings will continue to be held twice annually. This frequency has been reviewed and confirmed to be appropriate, by the member projects

The CPD programme is strengthened by the work of Task Groups and topical sessions at TAG meetings. It is expected these practices will continue with the mutual consent of TAG members. The Programme is always open to suggestions for additional innovative methods to further enhance the information exchange process.

Enquiries regarding membership and participation in the Programme should be directed to the NEA Secretariat.

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Appendix 1: Description of the CPD decommissioning projects

Note: the project descriptions appear as presented by the project personnel. Limited editing has been done to clear up some language translation issues but for the large part, the summaries appear as presented so a not to change the style. The summaries vary in length and detail depending on the maturity of the project and the amount of relevant information available.

A1.1 Reactor Projects in Progress

A1.2 Category 2 Reactor Projects (dormant)

A1.3 Category 3 Reactor Projects (complete)

A2.1 Fuel Facility Projects in Progress

A2.2 Category 2 Fuel Facility Projects (dormant) (none at this time)

A2.3 Category 3 Fuel Facility Projects (complete)

A3.1 Projects no longer in the Programme

A1.1 Reactor project in progress

A1.1.1 BR3 PWR Reactor, Belgium

Organisation

Belgium Nuclear Research Centre (SCK•CEN)

Reactor type and operation

Westinghouse PWR 40MWt, 1962-1989

Project summary

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 pressurized 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.). A second set of internals (Vulcain) had to be designed and implemented to perform these full-scale experiments and qualification tests (see below).

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 full system decontamination of the primary loop, including the pressure vessel. This operation was performed in collaboration with Siemens using the HP/CORD process. The thermal shield was then the first part of the internals to be dismantled. Its size reduction, targeting an evacuation in 400 liter drums, was performed remotely underwater, the water providing an adequate shielding to protect the operators from the high radiation field produced by the activated 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.

The objective of this operation was to compare the 3 techniques in terms of secondary waste generation, dose uptake, operating manpower and overall implementation costs. The mechanical cutting was subsequently retained for the upcoming operations. The remaining internals, presenting an even higher activity (up to 4 Ci/kg or 150 GBq/kg 60Co) due to their closer proximity to the reactor core, were further dismantled using a specifically designed remote controlled band saw and a circular saw. The same techniques were used to segment the second set of internals (Westinghouse). This operation was also implemented in the framework of the Research and Technological Development Programme of the European Commission. All the highly activated internal pieces were packaged in 400 liter drums, then conditioned by grouting with cement and stored at the Belgian interim storage facility for radioactive waste (Belgoprocess). The Reactor Pressure Vessel (RPV) was removed out of its cavity as a whole piece in the middle of 2000. The pressure vessel itself was later segmented underwater using mainly the circular saw and the band saw. For the dismantling of the neutron shield tank (NST), a new technique had to be chosen since the use of the existing tools (mechanical saws) was incompatible with the complexity of this structure. The primary technological solution considered for this component was the use of an abrasive water jet cutting tool (AWJ) in combination with a remote controlled MAESTRO arm (Cybernétix). The water jet cutting tool would also later be used for the segmentation of other large components like the RPV remaining parts (bottom and cover) and the pressurizer

In parallel with the dismantling and the size reduction of the reactor internals and the RPV, the electromechanical dismantling of various contaminated auxiliary circuits was going on. The BR3 fuel was evacuated to BP for dry interim storage in a Castor® cask late 2002.

To deal with the (metallic) important radioactive material fluxes, alternative evacuation routes (recycling within the nuclear industry, decontamination targeting unconditional or conditional clearance, decay storage) have been thoroughly investigated and specific material management tools (databases) have been developed and implemented. For the decontamination of stainless steel and carbon steel pieces an innovative "hard" chemical decontamination process, based on the use of Ce(IV) in sulfuric acid, has been developed in close collaboration with Framatome ANP: the so-called MEDOC® process (for MEtal Decontamination by Oxidation using Cerium). The industrial facility has been put in operation in 1999. Up to now, about 80% of the 60 t of stainless steel pieces treated in the facility could be unconditionally cleared and sold to a scrap dealer. Among the remaining material, more than 80% have a residual radioactivity lower than 1 Bq/g allowing clearing them after melting in a nuclear foundry. To deal with slightly contaminated metallic material of simple geometry, a wet sand blasting unit has also been built.

Considering the very satisfying decontamination results obtained, in batch configuration, on stainless steel pieces, the SCK•CEN considered modifying the facility to allow closed loop treatment of large components, in particular the SG and the PR. The closed loop decontamination of the SG was performed over 3 weeks in late 2002 and resulted in the conditional release of the tube bundle (residual 60Co activity < 1 Bq/g). Following the good results obtained on the SG, the pressurizer (2003) and the Fuel Transfer Tank (2005) were also successfully decontaminated with the MEDOC process in closed loop, with the solution being sprayed on the wall rather than filling completely the component. The first cuts of the SG were performed using the AWJ tool coupled with the Maestro robotic arm. However, considering the complexity and the poor efficiency of the tool for segmenting the tube bundle (only 4 tube layers could be segmented in a single pass), the 3 last cut, through outer and inner wall and including tube bundle, were performed with a tailor made diamond wire in wet conditions.

Besides an extensive testing and qualification programme dedicated to concrete decontamination and demolition techniques running from 1994 to 1998, several significant concrete cutting and dismantling operations have been achieved to create access to components, to allow handling and evacuation of large components or simply ease logistics and material management:

- Opening of the plant containment (diamond circular saw).
- Opening of the operating deck (diamond wire sawing), allowing handling of the SG.
- Opening of the High Level Liquid Waste tanks rooms (diamond wire sawing).
- Opening of the Fuel storage pond (dry diamond wire sawing) and removal of the FTT casing (diamond circular saw).

The R&D programme related to radioactive concrete decontamination and dismantling was carried out in collaboration with the CSTC (Centre Scientifique et Technique de la Construction) and included, among others, cold and hot testing of hydraulic Rock Breaker, demonstration of explosive demolition on a 1:1 scale mock-up of the bioshield and electro migration feasibility study.

In 2005, the BR3 decommissioning project has entered a new stage where the main concern is the centralization of services/utilities within the main building so that the electromechanical dismantling of peripheral buildings can be completed, detailed characterisation performed and the clean up of the civil engineering works started. The following main operations have been successfully completed;

- dismantling of purification circuits, including the demineralization columns,
- dismantling of the first High Level Liquid Waste (HLLW) tank;

- study and implementation of a new general ventilation system;
- refurbishment of the Fuel Storage Pond as a cutting workshop.

In the auxiliary building cellars, the clean up of the rooms (400 m²) formerly housing the purification circuits, revealed to be particularly challenging:

- the removal of a highly contaminated and deeply embedded piping network required a clever combination of various concrete sawing and dismantling techniques;
- floors of demineralization cells were deeply contaminated and required strenuous manual hammering work.

Other main dismantling operations carried out in the period 2006 – 2009 include:

- conventional demolition of the ventilation stack after free release;
- dismantling of the turbine concrete sole.

As far as the reactor building is concerned, dismantling activities focused on the cutting and removal of the neutron shield tank, the last highly activated component remaining. The segmentation operation is accomplished using the Maestro Arm and the AWJ tool in different phases:

- Phase I (completed end 2007): cutting of the inner wall of the NST and sorting of the plates according to their activation level (medium and low active levels).
- Phase II and III (completed end 2008): cutting and removal of reinforcement ribs, allowing access for further segmentation and removal of instrumentation tubing.
- Phase IV (shall be completed end 2010): Segmentation and removal of instrumentation tubes, radial ribs and upper circular ribs, i.e. the removal of all medium activated parts.

In 2008, the Belgian nuclear safety authorities finally granted the decommissioning license of the BR3 project (the work was so far performed under a modified operation license with regular update of plant status). The decommissioning license defines the final objective of the project as green field with a time planning constraint of 2020. On site Interim storage of radwaste is clearly excluded in the scope of the decommissioning license.

Since 2009, activities are focusing on ventilation building cleanup. An extensive characterisation campaign revealed that one of the main issues could be the clean up in the Refueling Channel Water Storage Tank (RCWST), where large cracks have been located on the external wall. In the upper levels (ventilation and extraction rooms) clean up operations will mainly consist in superficial decontamination of the walls and ceilings (removing of the paint with grit blasting or manual grinding) and shaving of the floors, where contamination is mainly originating from operation of cutting workshop and interim storage of radioactive material rather than from plant operation.

Once the segmentation of the NST is completed, further dismantling in the Plant container becomes possible and includes:

- Removal of Refueling Channel liner.
- Dismantling of the in-pool support structure for removable shielding elements;
- Removal of the remaining primary pipe parts embedded in the bioshield.

Considering the large volume of rubble to be expected and that on site interim storage is not permitted, the removal of the bioshield activated concrete (~100 m³) is postponed until the final disposal containers for LLW, the so called "monolith" containers, are qualified and made available by ONDRAF/NIRAS.

Project timescale and cost estimates

1990 – 2020, 150M Euros.

Unique attributes, challenges

Special design features of the BR3 reactor include:

- Primary circuit configuration = 1,5 loop (1 hot leg, 2 cold legs, 2 primary pump, 1 SG),
- 1 m large Neutron Shield Tank around the RPV
- 2 sets of internals were successively used during reactor operation.

The development and the implementation of an internal tube cutting machine to separate the RPV (from primary pipe penetration) was particularly challenging due to the small inner diameter of the pipe (with respect to the pipe thickness).

New methodology developments

- The in-house developed Material Management Database allowed integrating the material production process in a QA programme (ISO 9002 certification).
- Non destructive assessment of contaminants (Cs-137) penetration depth in civil engineering works.
- Abrasive Water Jet cutting remotely operated by robotic arm (MAESTRO).
- Closed loop chemical decontamination of primary circuit large components targeting their (conditional) clearance and/or decategorization.
- A dedicated database for the clean up of civil engineering works ensuring all time availability and adequate archiving of the huge amount of data generated by the clean up process (and in particular the radiological measurements).
- Segmentation of a Steam Generator.
- Dry cutting of reinforced concrete with diamond wire.

New equipment, instrumentation developments

The BR3 decommissioning project was selected by the EC as pilot project for the assessment of dismantling techniques and decommissioning costs. In the framework of this research contract, one of the most important items was the comparison of methodologies and techniques for the segmentation of highly active internals: thus plasma arc torch, electric discharge machining (or sparking erosion) and the mechanical milling were tested.

The following equipment were developed in the framework of the project:

- ⇨ A custom made band saw with rotating blade (Rolls Royce).
- ⇨ The hard chemical decontamination process based on the use of Ce(IV), the so called MEDOC process, for the free release of contaminated metals.

Licensing, political issues

BR3 Decommissioning planning extended due to unavailability of definitive containers for final disposal as LLW (bottleneck for the evacuation of the 150 m^3 of activated concrete).

Lessons learned

The development and implementation of an adequate panel of decontamination techniques (mechanical, physical, chemical) allow investigating alternative evacuation routes for material arising from the dismantling, such as unconditional clearance, conditional clearance (after melting), recycling for nuclear application. The BR3 experience demonstrates that decontamination can be implemented in a cost effective way and leads to a significant reduction of radwaste volume and subsequently to a better control of decommissioning cost.

Implementation of a good material management system (in a QA framework) is compulsory to deal with important material fluxes, prevent bottleneck, optimize material evacuation routes and satisfy with the requirements of clearance practices.

A larger capacity for on site Interim storage is necessary to limit bottlenecks in material management streams and avoid interim storage inside the plant.

Availability of a hot cutting workshop from the start of the dismantling project, placed at a logical area. Eventually enlargement of the controlled area for material treatment should be considered.

A1.1.2 Whiteshell Research Laboratory (WRL) Decommissioning, AECL, Canada

The AECL Whiteshell Laboratories (WL) is located in Pinawa, Manitoba, Canada (about 100 km NE of Winnipeg in central Canada). The site has operated since 1964 and features:

- An organic cooled reactor (WR-1)
- Shielded facilities, hot cells and radiochemical labs
- Waste storage facilities
- Infrastructure support systems (offices, workshops, power house, labs, etc)

The Site hosted a variety of activities including:

- CANada Deuterium Uranium (CANDU™) Reactor Research and Development
- Reactor safety programmes.
- Nuclear fuel waste management programme.
- Small reactor development
- Environmental research.
- Chemical and material science programs.

The two primary nuclear facilities on site are the WR-1 Reactor and the Building 300 (B300) Radioisotope Laboratory Complex. The WR-1 Reactor was organic cooled, heavy water moderated and was used to test the use of organic coolants and for the development of advanced CANDU fuels. The reactor operated from 1966 to1986. The reactor operated up to 60 MWth. There was no electrical generation although in the latter years it was utilized to supplement site heating. The reactor was decommissioned to its current state of Storage with Surveillance (SWS) in 1996.

The B300 complex housed a variety of radiological and non radiological facilities including:

- Shielded Facilities (SF), Hot Cell Facilities (HCF), and Immobilized Fuel Test Facilities (IFTF).
- Various mechanical and chemical laboratories.
- Van de Graaff Accelerator.
- Neutron Generator.
- Thermalhydraulic Experimental Loops.
- Machine shops.
- Offices, conference rooms and the computer centre.

Shutdown operations and site cleanup has been ongoing since the late 1990s when the site closure decision was announced. In April, 2002, the Environmental Assessment (EA) "Whiteshell Laboratories Decommissioning Project Comprehensive Study Report" was approved and subsequently the site decommissioning licence was issued in January, 2003. The licence was valid for 6 years, and was subsequently renewed for an additional 10 years (currently expires December, 2018).

The final end state will be "Green Field" condition, achieved through several stages. Most of the Site infrastructure buildings, with the exception of the reactor and some enabling facilities, will be dismantled in the next couple decades. The reactor is currently scheduled to be dismantled circa 2030. Waste storage facilities will remain operational until a final national waste repository is established and operational (c. 2040).

Unique attributes/challenges

The WL Waste Management Area (WMA) contains a "standpipes" area. There are 171 standpipes. There are sixty-nine standpipes that contain fissionable (fuel) materials, many of which are mixed with other intermediate-level waste (e.g., filters, scrap metal materials), with the likelihood that they may contain some flammable gases and in some cases, pyrophoric fuel materials. These sixty-nine standpipes may be remediated.

New methodology developments

None yet, though there may be some for the planned remediation of the WMA.

New equipment or instrumentation developments

None yet, though there may be some for the planned remediation of the WMA.

Licensing and political issues

The Environmental Assessment (EA) "Whiteshell Laboratories Decommissioning Project Comprehensive Study Report" was approved by the regulator (Canadian Nuclear Safety Commission (CNSC)) in 2002. The EA covers the entire site decommissioning.

The WL Decommissioning is carried out under a strict Quality Assurance programme – Canadian Standards Association N286.6. The CNSC approved the WL Decommissioning Quality Assurance Plan in 2007.

Lessons learned

None yet that are significant.

A1.1.3 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. This strategy was subsequently changed to decommission the facility to Stage 3. A decree for total dismantling (stage 3) was obtained in February 2006 but cancelled 6th June 2007 by council of state for insufficient public information. Application for a new decree with public enquiry is underway.

Stage 2 has been achieved. Between 2002 and 2007 the following non nuclear buildings were demolished:

- Spent fuel storage building.
- Auxiliary building.
- Office building.
- Life base building.
- Liquid waste treatment building.

During the achievement of Stage 2

- Some 1 500 t of M and or LLW and 5 500 t of VLLW was produced.
- 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 2mm of concrete (11 000 m^2).
- Category 2: Surface with suspected liquid superficial radioactive contamination. Removal of a maximum of 6mm 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)

In total 730 000 hours were worked. The total dose received vas 115.5 mSv (0.3 mSv/year/worker).

There were some issues surrounding the demolition of the liquid waste treatment building. In 2004, the demolition of the upper part of the building was accomplished. It was then discovered that the concrete lower part of the building was contaminated to higher levels than expected due to leakage from liquid waste drums. Due to decontamination efforts (concrete removal) the work had to be stopped in December 2005 due to weakening of the structure. In preparation for final dismantling the following work has been undertaken:

- The reactor building has been refitted in 2005 : access, locker rooms, ventilation, crane, radioactivity monitoring equipment
- The temporary waste storage building has been finished (ex turbine building)
- The dismantlement of the reactor building, except C02-D20, heat exchangers has begun in 2005 and is actually finished.

Work has been stopped since 2007 awaiting a new decree.

A1.1.4 Bugey 1, France

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

- The reactor is in a pre-stressed concrete pressure vessel with internal dimensions 17 m diameter, 40 m high and external dimensions 28 m diameter and 56 m high.
- The reactor pressure vessel contains the core of 2 600 t of graphite (15 m diameter, 9 m high) above the heat exchangers that are integrated into the vessel.
- The vessel internals include 942 steel guide tubes above the core, the “corset” around the graphite core, the 15 780 hexagonal graphite bricks making up the core, the hot plenum of the gases exiting the core as well as the heat exchangers on their metallic cylindrical support.

The current status is as follows:

- Fuel has been removed.
- Control rods are still in place.
- High activity waste is temporarily stored in the upper slab cavities.
- Coolant turbines, power generating and auxiliary equipment have been removed.
- The pump station has been demolished (2003-2004).
- Caisson inspections have been performed in 2003 and 2009.

Basic design studies, based on proposals from experienced nuclear decommissioning contractors, have been performed by EDF – CIDEN (Centre d’ingénierie, déconstruction et environnement) resulting in the choice of an underwater scenario. The reasons for this are several, and include i) concerns from the French regulators that a fire involving graphite fines could not be totally ruled out, ii) that the shielding afforded by the water would enable a more “hands on” approach to be adopted rather than a fully remote operation as was the case at WAGR, and iii) greater flexibility to respond to unforeseen technical problems during dismantling, seen as particularly important bearing in mind the size and complexity of the Bugey reactor.

The reactor dismantling scenario is based on the main following stages:

- Preliminary works in order to make the caisson watertight.
- Top Cap opening.
- Upper internals removal.
- Sidewall concrete removal.
- Core support dismantling.
- Water draining within the caisson.
- Lower internals dismantling with heat exchangers.
- Final decontamination of concrete.
- Concrete structural demolition and site restoration.

Based on this scenario, a decommissioning licence has been issued in November 2008 which allows EDF to start Bugey 1 reactor dismantling with the objective to have an underwater caisson in year 2016.

Prior to reactor dismantling, ventilation systems improvements and new gaseous measurements set-up regarding environmental issues, have been performed in 2009 to allow the final dismantling of the electromechanical equipment and the removal of operational waste. These latter operations have been contracted and the detailed studies are currently under progress that will allow the beginning of the first operations on site by the end of 2010 for a total duration of 2 years.

The next issues to be contracted are the water treatment system, the new ventilation system and the waste route facilities.

A1.1.5 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 15 m, height 9 m, pond walls thickness 0.8 m). It was dedicated to fundamental research, technological and materials irradiations and production of radioisotopes.

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

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

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

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

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

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

Due to unexpected structure activation on the front block of the reactor, heavy deconstruction was undertaken in 2007. The same year site radiological mapping was performed.

Final dismantling and clean-up operation occurred in 2008-2009. The final radiological mapping of the buildings was performed at the end of 2009.

The Delicensing (declassification of the facility) was obtained in December 2009.

Total cost of the project is estimated to be 27.7M Euros.

A1.1.6 Phenix, France

Phenix is the only remaining operational French fast breeder reactor after the shutdown of SuperPhenix (1999). The reactor is located at the Marcoule site and belongs to EDF and the French Atomic Energy Commission (CEA). The reactor was developed at the end of the 1960s and has been in operation 1973. Initially the reactor operated as a power plant (250 MW electrical) as a demonstration reactor for sodium cooled FBRs. Since 1991, the reactor has been used as an irradiation tool for the actinide transmutation programme. The reactor underwent a major refurbishment in from 1999 to 2003. Final shutdown of the reactor was realized in late 2009.

This is one of the few reactors that have started active planning for decommissioning prior to shutdown. It was decided to begin the dismantling phase without waiting for a decay period because it was felt (from past experience) that there are no economic or biological benefits to this approach. As well, the experienced operating staff will be available to participate in the D&D phase. This prompt approach will require storage facilities for waste shortly after shutdown. National facilities are available for low and intermediate level waste. A new high level waste storage facility will be available at Marcoule around 2012.

The Phenix facility is comprised of four main buildings:

- Handling building, B 302. Contains the fuel dismantling line for examination of experimental fuel. Also contains the maintenance line for removable components and the intermediate storage tank for fuel decay.
- Reactor Building, B 301. Contains the reactor vessel and the primary sodium purification loop as well as part of 3 secondary loops.
- Steam Generator Building, B 304. Contains 3 secondary loops which transfer power to the water in the steam generators. The secondary loops have been shut down since the refurbishment.
- Turbo alternator building, B305.

The end of life tests (for the next generation research) were performed in 2009 with an international participation in the frame work of IAEA's Coordinated Research Project (CRP) and bilateral collaborations.

The preparation for dismantling also started in 2009 with the following objectives:

- reduce risk for subsequent operations (e.g. draining and rinsing of lubrication circuit, draining sodium circuits, etc.);
- remove "source terms": unloading sub-assemblies, removable reactor components, etc.;

- simplify and adapt parts of the facility;
- prepare premises and set up equipment that will be required for the dismantling phase.

The main objective of the Phenix dismantling project is to eliminate all the process equipment to clean the buildings to a conventional state. To achieve this objective, three main hazards must be eliminated.

- Fuel. Removal, handling and processing poses criticality hazards. Fuel will be sent to La Hague for processing. High level waste will be stored at Marcoule.
- Sodium. All sodium inventories will be treated at a new facility to be constructed at Marcoule.
- Radioactive equipment. High activation due to the presence of cobalt alloys.

Two main sodium treatment facilities will be built (and later dismantled):

- NOAH (bulk liquid sodium).
- ELA (active sodium solid waste).

These facilities will start in 2016.

Because of knowledge already gained from other projects, these hazards do not pose any unmanageable technical challenges.

The overall master plan foresees the preparatory activities to take place between the period 2009 to 2013 and the Dismantling Phase to occur from 2013 to 2023. The overall budget best estimate (including singular sodium waste treatment from other sites of CEA) is €750M, with an uncertainty of - €50M and + €350M.

A1.1.7 MZFR, Germany

The MZFR was a 200 MWth (57 MWe) pressurised heavy water reactor operated by the former Kernforschungszentrum Karlsruhe from 1966 to 1984. The plant is being decommissioned to a Stage 3 status under eight sub-licences.

The complete dismantling of the reactor will require eight dismantling steps in total. For every dismantling step, a separate dismantling license had to be applied for. Since early 2007, all licenses needed for MZFR decommissioning to the green field under the Atomic Energy Act have been granted.

Work under the first seven licences has been completed. Currently, the activated inner layer of the biological shield is being dismantled under sub-licence number eight.

During earlier work, all D2O systems were removed, the systems dried, the cooling towers demolished, and the water treatment plant dismantled. The turbine hall was cleared and is still being used as 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 15 was obtained. Another important activity under the same sub-licence was the dismantling of the 4 D2O enrichment columns, each 12 m in height and 1.5 m in diameter.

The safeguards facilities on site were removed and the security fence taken down under the fifth sub-licence.

The criteria for release of components from the plant are that the surface activity should be less than 0.5 Bq/cm2. Release is accomplished 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 to the reactor building. The caisson is large enough to allow 20-foot containers, weighing up to 20 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 has been established with its own ventilation system.

The 6th licence mainly covered 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 into the reactor building, etc. During this work, the two 55-tonne steam generators, the 20 tonne 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.

Licence Seven – Steps of dismantling of the reactor pressure vessel and internals

First step **-** *The one piece removal of the rod-shaped components*

The removal of the rod-shaped components started in April 2000 and was finished in September 2000. The dismantling of the reactor vessel internals (cooling channels, absorber rods, leading rods) under licence seven was complicated. There were 121 vertical coolant channels for fuel assemblies (without fuel) and 18 control rod absorbers in 18 guide tubes that were located at an angle of 20° to the vertical. Outside of the vessel head, a drive and position indication tube were located for each absorber. Five cooling channels could not be removed and therefore they have been cut above the moderator tank and cut in step three - wet dismantling.

The cartridges with the fuel elements were transferred in the transport device to the waste treatment department (HDB) in Karlsruhe for dismantling in a hot cell.

Second step

The equipment for dry dismantling of the RPV components (lid, upper spacer, lower spacer, RPV) was installed in 2001.

Divided into two phases - dry dismantling part 1 and part 2, the second step comprised the disassembly of the RPV bolt and RPV sockets and the dismantling of the RPV lid as well as the upper spacer and its weight ring and which has also been finished.

The band saw was tested in the installed state at MZFR and started cutting of the RPV lid in April 2002. The second step was finished in summer 2003

Third step- Dismantling of the moderator tank, the five remaining cooling channels and the thermal shield

The equipment for wet dismantling (cutting under water) of the moderator tank and the thermal shield was manufactured, tested, and optimised in an external test bed (mock up VAK) from 2001 until 2004. In addition, the plasma-cutting device was tested at the University of Hanover.

Thermal and mechanical dismantling procedures were applied to dismantle the moderator tank and the five absorber rods, which could not be taken out during the dismantling of the rod-shaped components.

The equipment was constructed for all known cutting tasks and tested at the test facilities. The above-mentioned plasma cutting torch was especially developed for under water application with swirl gas technology for the automated CNC-controlled cutting of super alloyed stainless steel. This special torch-exchange head has been developed in order to optimise the exchange of wear and tear elements and therefore to minimise the dose for the dismantling team.

The separation of the five remaining cooling tubes was carried out with hydraulic shears, which was controlled by a hydraulic under-water master-slave manipulator. All applied devices were operated and controlled from the control room outside of the containment.

The transport basket, which was filled with separation tube parts was transported via a basket gripper out of the RPV and placed into a contamination-safe barrel, dried in an electrical drying facility and afterwards loaded into a repository cask.

The dismantling of the moderator tank with deflectors with a material thickness up to 50 mm was accomplished using the plasma cutting unit due to the high activity under water.

The dismantling of the thermal shield with a material thickness up to 130 mm has been effected with the before explained plasma cutting unit under water.

The used plasma cutting unit was performed to dismantle the whole Thermal Shield. Because of the small gap between RPV and Thermal Shield within 14 mm it was impossible to dismantle all five Thermal Shield rings with the plasma cutting unit. The 1st until the 4th ring within a material thickness of 70 mm was removable in time.

The 5th ring from 70 mm up to 130 mm could not be removed by the plasma cutting system because of the produced slag which would weld thermal shield parts with the RPV. The proof was made by a test-cut with plasma-cutting-system.

To solve this problem the choice was the CAMC (Contact Arc Metal Cutting)-System. The CAMC-Cutting system based on the well known high current cutting process and has to be qualified in a very short time for this task.

The CAMC-Cutting system was as well as the Plasma-Cutting system especially developed for the under water dismantling. The cutting head was adapted to the already existing five-axis die carrier. The same CNC-controlling unit was used to move the axes of the die carrier with a maximum accuracy.

Wet dismantling of the moderator tank and thermal shield by using a plasma cutting unit and the necessary devices in up to 8 m water depth has also been finished in November 2005

Last step - Removing and dismantling of the lower filler piece and the Pressure Vessel

The lower filler piece has been cut with the band saw. After the vessel was lifted off the RPV-insulation was removed. For cutting the RPV flange (1st segm.) the Band Saw was used. The main dismantling equipment was a flame cutting unit.

The removal of the RPV and its internals was successfully completed in 2007.

License Eight – 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 cleanup of the site

Besides the demolition of the building structure, the major task that still remains to be accomplished is the remote dismantling of the activated part of the biological shield. The various cutting tools and an adapted conventional excavator as a carrier system are being tested. Operation under active conditions was started in spring of the year 2010.

The remaining activated components are located in the activated part of the biological shield. These components include the reinforcement and a so-called steel liner that served as a self-supporting permanent sheathing in the construction phase. In addition, 12 measurement chambers are installed radially in the biological shield.

The equipment to be used for dismantling the activated inner part of the biological shield was put up for testing in the machine hall of the MZFR. A company has been contracted to execute the tests and subsequent dismantling of the biological shield.

Dismantling Concept

Remote dismantling of the activated biological shield takes place in three steps that have to be executed successively:

- Phase 1: RC dismantling of the steel liner and cutting of the inner reinforcement layer.

- Phase 2: RC dismantling of the inner layer of the activated concrete of the bioshield (hanging dismantl. rack).
- Phase 3: RC dismantling of the inner layer of the activated concrete of the bioshield (dismantl. rack in the standing position).

All phases are executed using the same dismantling, transportation, handling, and auxiliary equipment. It will be presented in the next section.

Dismantling equipment installed in the test bed

The activated part of the biological shield is dismantled using a remotely controlled excavator as tool carrier system. It is positioned in the biological shield on a special excavator platform, the hanging and standing rack.

To test the dismantling equipment in the MZFR machine hall, a representative part of the biological shield was rebuilt in original size (dummy). In this way, comparability with real dismantling conditions was ensured.

The tests cover all systems and components listed below, which are required for real dismantling:

- A rotating movable carrier ring, from which a hanging rack is suspended as a working platform for the demolition excavator;
- an electrohydraulic demolition excavator, to which various standardized and special dismantling tools can be attached by a rapid-exchange coupling system;
- a remotely controlled master-slave manipulator for supporting work, which can be moved vertically on the hanging rack;
- peripheral devices fixed to the hanging rack, including the surface cleaning device, ultrasonic detection system with marking units, etc.;
- and the complete audio/video system, including all control panels and control units required.

The demolition excavator, manipulator, and all standardized and special tools were tested in the combined mode. This also allowed for an optimization of the dismantling strategy.

The operation staff was trained extensively on the MZFR test rig using all systems and components, including the demolition excavator and the manipulator. The dummy biological shield was dismantled remotely and successively under conditions comparable to later hot dismantling.

Based on this test operation and the experience gained, the equipment and procedures were optimized and qualified.

The major technical objective, i.e. cutting of the steel liner and the inner reinforcing layer, including the inner concrete layer, in a single process was completed successfully.

Installation of the Equipment in the controlled area

After about six months of test operation and acceptance by an inspector present at the site, the equipment was moved to the reactor building in autumn 2009.

In the controlled area, the first systems and devices needed has been commissioned in the presence of the inspector in December 2009.

Phase 1, the remote controlled dismantling of the steel liner and cutting of the inner reinforcement layer was successful finished in May 2010.

In July 2010, the RC dismantling of the inner layer of the activated biological shield was started.

Summary

With its decision to completely dismantle the MZFR, the former Kernforschungszentrum Karlsruhe entered a terrain that was completely unknown at that time. From the very beginning, maintenance of competence of own and external personnel has been of particular relevance to the execution of this dismantling project.

Since July 2009 the WAK GmbH is responsible for all dismantling activities for decommissioned nuclear test and prototype plants and for the treatment of radioactive wastes on the site.

As a highlight visible from outside, the 99.5 m high exhaust air stack was shortened to +18.5 m in October 2009, see figure 4. Dismantling of this pure steel stack gave rise to about 50 tonnes of steel, which were checked for the absence of contamination and then transferred to conventional reuse.

And today?

The last activated components of the MZFR will be removed when the biological shield will be dismantled remotely. This work was started in July 2010. The first phase, dismantling of the steel liner, was finished. This step is aimed at dismantling the liner that determines the dose rate and the inner reinforcing layer.

Phases 2 and 3 will cover concrete dismantling of the activated biological shield. This work will be also controlled remotely and take about 12 months.

When the biological shield is dismantled, infrastructure facilities will be removed successively. Then, all controlled areas will be decontaminated. Remaining work at the MZFR will include the extensive decontamination of all radiation protection areas and the removal of the H3-contaminated concrete structures.

Formation of a permanent team of regional expert planners has proved its worth during MZFR dismantling. Effective construction, dismantling, and operations management in cooperation with these expert planners allows for an efficient execution of the tasks.

The dismantling techniques applied are based on industry-tested methods of increased availability and optimized spare part management. These methods are complemented by special solutions for the dismantling of highly activated reactor internals and components. Both the plasma cutting facility and the CAMC technology used are unique in terms of performance.

Extensive processing and testing workings precede the above described dismantling steps; a fundamental precondition for the successful realisation of complex dismantling operations nuclear power engineering.

Delays of decommissioning work have been minimized by risk studies and intensive testing.

Close cooperation with experts and the authorities as regards the decommissioning and remaining operation had a very positive effect on the execution of the project.

A1.1.8 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. Energie-werke Nord GmbH was created to decommission these plants in a socially acceptable form.

Four of the eight Greifswald reactors (WWER 230) had been in extended operation between 1973-1990. The fifth, which was of a more recent design (WWER 213), had been started in 1989 and was only a short trial phase into operation. Unit 6 was ready for operation, while the other two units (7,8) 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, to use for the decommissioning. 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 material inside the buildings including all contaminated and activated systems and treatment and conditioning of the dismantled material with the aim of free release the materials so much as reasonable possible.
- The site restoration phase and remaining operation phase comprises: dismantling of remaining systems, building decontamination and demolition, the restoration or adaptation of the site for other uses and the operation of all remaining facilities including care and maintenance of new facilities erected in the frame of decommissioning, like the Interim Storage North and the reconstructed Warm Workshop.
- Currently, the project is in the dismantling phase and site restoration phase. Both phases are running in parallel to allow an early site reuse for different industrial purposes. A special agreement with the authorities has been established for a step by step free release procedure of the site.

The main project issues are described below:

- EWN is faced with one of the largest decommissioning project world wide. The reason is the typical Eastern construction of the reactor units, as described above. A huge material mass has to be dismantled (1 800 000 tonnes) on the Greifswald site.
- A central feature of the decommissioning strategy at Greifswald is the Interim Storage Facility North (ISN), which allows the cutting of large components within the facility for later treatment when convenient. The ISN will also house fuel elements. The operation had to be suspended briefly due to the lack of a licence (under Article 37) from the European Commission. In September 1999, the operation was restarted. The ISN has 8 halls, each with a storage capacity of 25 000 m^3.

  In hall 8 are stored 65 CASTOR casks with all spent fuel from the Greifswald and Rheinsberg sites.

  In hall 7 are stored all large components such as steam generators for decay and step by step cutting by the big band saw, also located in the ISN. The reactor internals of the Greifswald reactors 3,4,5 and the Rheinsberg reactor are also stored in ISN as complete components in special Shielding and Transport Devices (STD).

  Furthermore the ISN made it possible to uncouple the material treatment from the dismantling work on spot and in so far that was the basis for a smoothly running waste (material) management in parallel to the dismantling on spot.
- Under the current licence, all fuel elements were transferred out of the wet on-site storage into dry storage in CASTOR casks by the end of 2004. In order to begin dismantling works under easier safety restrictions, all fuel elements at the various reactor units were transferred into the wet storage. Thereafter, all spent fuel was put in special designed CASTOR casks (CASTOR 440/84) – each CASTOR cask to hold 84 Russian spent fuel elements. That reloading work was done in a new constructed reloading station, using the former spent fuel pond of unit 3. All spent fuel is now (since mid-2006) stored in the ISN.
- 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.

- The remote dismantling was foreseen for the reactors and the internals of Units 1-5 of the Greifswald plant. Prior to this the equipment to be used was successful tested in Unit 5, using the non activated RPV and the internals of units 6-8, which were never in operation.

  After transport of the RPV 5 into the ISN hall 7 in December 2003 and additional in situ radiological measurements were done on the reactor 1, which was the longest time in operation, an evaluation was made of two possible dismantling strategies: the in-situ remote dismantling (cutting and packaging) of activated components versus removal and transport as a large complete unit for storage and decay into the ISN. The results were:

  – The execution of both dismantling strategies was successful.

  – The facilities and the equipment are operational and reliable.

  – All sequences of operations were tested and optimized.

  But on the basis of following evaluation results (table below) the dismantling strategy was changed from remote dismantling (cutting in packing size) to large component management:

	Cutting in packing size		Large components management
Project time	Engineering	100%	45 to 65%
	Preparation	100%	35 to 55%
	Execution	100%	8 to 12%
Waste management	Stor. Volume	100%	50 to 70%
Collective dose rate		100%	15 to 20%
Costs		100%	40 to 55%

- To date a total mass of 230 000 tonnes of material has been dismantled at Greifswald. Dismantling in the controlled areas unit 1-5 is approximately 77% complete while in the supervised area of unit 1-5 (turbine hall) approximately 99% of all material has been dismantled.

  In the ISN 24,400 Mg material/waste have been stored including:

  – 5 reactor pressure vessel (~214 Mg) + 1 RPV from KKR,

  – 18 steam generators (~160 Mg), and

  – 4 pressurizers (107 Mg).

- There has been a drastic reduction in the work force at Greifswald over the years (from 5 000 to now 950 workers). This has been achieved by retirement schemes, privatisation of technical and service work, and some unavoidable dismissals.

  On the other hand, due to the early implementation of a special site reuse project including the strong encouragement of new smaller companies to locate on the former NPP site, up to 700 new employee positions could be established. That strategic issue had a positive influence on the decommissioning process.

A1.1.9 Arbeitsgemeinschaft Versuchsreaktor (AVR), Germany

The 15 MWe AVR reactor in the direct neighbourhood of the Julich Research Centre was a high-temperature, helium-cooled reactor with spherical fuel (pebble bed) developed in Germany. This research reactor operated between 1967 and 1988. Besides the execution of physical experiments, it was used as a test bed for different types of fuel elements.

The licence to transform the plant to safe enclosure conditions was granted in March 1994 under the short term 7/15. Under this license and as a first dismantling phase the following work was performed:

- Defuelling (1994-1998) of the reactor and removal of some 100 000 spherical fuel elements off-site. Interim storage in special CASTOR casks at Juelich Research Centre.
- Dismantling of systems in the turbine hall and outside the reactor building, e.g. components of the secondary and main cooling-water loops.
- Dismantling of the three cooling towers.

Between 1998 and 2002 license 7/15 was complemented by four amendments. This second phase included the following activities:

- Inspection of the core cavity (by video camera) and of the refuelling system for residual fuel.
- Dismantling of the helium supply system.
- Dismantling of systems in the annular auxiliary buildings: liquid nitrogen generation components, compressed gas cylinders of the helium supply system, safety relevant fittings of the secondary loop e. g. the emergency shutdown valves, the main isolating valve and many parts of the control systems.
- Dismantling of systems and components in the confinement e.g. cooling-water loops, refuelling system, coolant gas circulators as well as the steam generator loop in the upper part of the confinement.

In 2003 the AVR company was taken over by Energiewerke Nord GmbH (EWN). In the course of this acquisition the project objectives were revised to direct complete dismantling. In order to realize this; the reactor pressure vessel (RPV) had to be grouted before lifting it out of the reactor building as one part. Afterwards it will be transported to an interim storage facility where it will be dismantled in 30 to 60 years. The rest of the plant will be completely dismantled. Contaminated soil under the reactor building will be remediated to green field conditions.

After the acquisition by EWN decommissioning activities were carried out under the fifth amendment of license 7/15 which was granted in 2004. Under this amendment a material lock building adjacent to and enclosing the top of the reactor building - to provide a new powerful transportation route out of the confinement, and in particular for the later removal of the complete RPV – was constructed.

Inside this material lock building the top of the old reactor building was removed, i.e. the roof and the water tanks standing on top of the biological shield. A first temporary closing system (closing system 1) with hatch function on top of the biological shield was completed in June 2008. It now forms the top part of the confinement.

In November 2008 the RPV was grouted with a low density cellular concrete in order to achieve:

- minimization of radioactivity release by air plane crash;
- stabilisation of the reactor-internals during removal of the vessel;
- fixation of graphite dust and thus minimizing radioactivity release in case of handling accidents.

The low density cellular concrete (density 0.7 g/cm^3) consists of blast-furnace cement with low hydration-heat and low chloride content mixed with foam glass spheres (diameter 0.1 mm and 0.2 mm).

After the grouting and under protection of closing system 1, the top of the confinement vessel was dismantled by cutting out single segments. By construction of closing system 1 and opening the confinement the new transportation route out of the confinement was ready for use.

On 31st March 2009 the regulatory authority granted the license for green field decommissioning of the AVR reactor site.

During the year 2009 the work was focused on:

- removal of infrastructure used for the grouting of the RPV;
- dismantling of the helium purification plant;
- further dismantling of working platforms in the confinement;
- preparatory work for the handling- and transportation system (see below);
- construction of the interim storage facility for the RPV.

The construction license for the interim storage facility was granted in 2007 and the operating licence in March 2010. The new storage hall V for the dismantled material is in operation.

Future programme and time schedule

After the grouting of the RPV the next important project task will be its removal into the material lock building. For this purpose a handling- and transportation system has to be designed, constructed and installed. It consists of

- an upper lifting tool at the reactor vessel;
- a slide-system with strand jacks for lifting and removing the reactor pressure vessel into the material lock building;
- a rack (support) for mounting the lower lifting tool; and
- a transportation skid.

Construction and installation of the handling- and transportation system will be carried out in 2010 and 2011. The transport of the reactor pressure vessel is scheduled for early 2012.

The last project phase after removal of the reactor pressure vessel will involve the following activities:

- dismantling of remaining components in the confinement;
- dismantling of concrete structures in the confinement;
- dismantling of all installations inside the buildings;
- release of building- and concrete structures from regulatory control;
- demolishing of building structures;
- release of the AVR site from regulatory control;

The end of the project is scheduled for 2015.

Unique attributes/challenges

- High-temperature, helium-cooled reactor with spherical fuel (pebble bed).
- Heavy contamination of notably Sr90 inside reactor pressure vessel, see table below:

Total fission product activities		Total activation products	
Cs137:	3.0 E13Bq	C14:	3.0 E14Bq
Sr90:	5.0 E13Bq	Co60:	1.5 E14Bq
		H3:	3.0 E15Bq

- Slight contamination with Sr90 of ground around and under reactor building in consequence of a steam generator incident in 1978.
- Due to early mobilized carbon dust and H3 all dismantling work connected to reactor pressure vessel must be performed with air supplied suits.

- Expected amounts of dismantled material for green field decommissioning:

Reactor pressure vessel:	2 003 Mg
Metals:	4 300 Mg (thereof free-releasable 3 870 Mg)
Rubble, excavated soil,	
non-metals:	32 200 Mg (thereof free-releasable 30 600 Mg)
Other wastes:	115 Mg

New methodology developments

- Removal of the RPV with a weight of approx. 2 000 Mg

New equipment or instrumentation developments

- Fast and economic patent-registered measuring method for determination Sr90-acitivity concentrations in concrete and soil down to a detection limit of 40 Bq/kg

Licensing and political issues

- Revision of project objectives after takeover of AVR by EWN and performance of a new licensing procedure for green field decommissioning

Lessons learned

- Employ people instead of using contractors for the execution of the decommissioning works

A1.1.10 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, N2 cooling system, the turbine and steam generator halls, etc. The seventh sub-licence was to prepare the dismantling of the primary systems, while the actual dismantling of those systems is taking place under the eighth sub-licence. Ongoing work is concentrated to the reactor building and the dismantling of the primary cell and sodium-cleaning cell has been completed. Dismantling of the reactor internals is ongoing (mid 2006). Remaining quantities of sodium call for special attention e.g. washing under inert (Nitrogen) atmosphere and the use of cold cutting techniques.

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

Work continues under the 9^{th} licence (step 9), package #1 (dismantle reactor tank). This work, mostly completed, includes:

- Cable guided cladding tubes
- Streaming cup
- Thermo-sleeve of sodium-entry pipe
- Lower thermo-shock panel
- Reactor vessel internals

Preparatory work is underway in anticipation of the commencement of Package 2 (primary shielding, heat insulation, biological shield and internals). The cast iron blocks for the primary shielding mock-up tests have been manufactured and factory test have been completed. The lifting tool for the primary shielding has been manufactured and factory acceptance tests are complete.

KNK was owned by FZK but was operated by the company KBG, which was a subsidiary of the local utilities. Meanwhile, the services of the operational company KBG has been terminated. Since July 2009 the WAK GmbH is responsible for all dismantling activities for decommissioned nuclear test and prototype plants and for the treatment of radioactive wastes on the site.

The project is now scheduled to be completed in 2011 at a cost of €309.2M. This is slightly over 2005 projections.

A1.1.11 Garigliano, Italy

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

At the Garigliano site:

- Spent fuel has been shipped off site.
- Safe enclosure of reactor and turbine buildings was reached in 1998. The reactor building containment has been isolated from the other buildings and ventilated by utilising the temperature and pressure differences between the inside and outside of the containment.
- Radioactive waste treated at site consists of:
  - LLW: 796 packages (320 litres) of super-compacted Dry Active Waste (DAWs), previously stored in 2 429 drums (220 litres), 86 packages (320 litres) of non-compressible DAWs, after sorting and monitoring.
  - ILW: 280 m^3 of sludge, 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 5×109 ITL (approximately 2,5 MEuro).
  - HLW: 4 t of highly activated materials (fuel channels, control rods, in-core parts, etc.) were retrieved from the storage trench and cemented in 6 concrete containers (50 t, 15 m^3 each).

- Other LLW waste is stored at plant site.
  - 600 t of material has been released from the controlled area at a release limit of 1 Bq/g or cm^2 (β/γ) or 0.1 Bq/g or cm^2 (α).
  - Chemical and mechanical decontamination methods have been tested for stainless and carbon steel large tanks.

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

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

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

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

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

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

The outstanding priorities are:

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

As of 2010, a national repository for waste has not been realised. Recent work has focussed on demolition of conventional buildings to make room for a waste storage facility and removal of asbestos from the rector building. Construction of the waste storage facility has started but is hampered with contractual problems.

A1.1.12 Latina, Italy

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

At the Latina site:

- Spent fuel has been shipped off site.
- Some preliminary decommissioning activities began in 1992 and concerned the dismantling of some systems and components no longer safety-related, such as:
  - Water-steam piping and auxiliary piping.
  - Thermal insulation of boilers and primary circuit ducts.
  - Biological shield fans.
  - Fuel charge/discharge machines.
  - CO2 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.

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. At present, work has halted at the Latina sight due to lack of waste storage facilities. Work is now focussed on the construction of a LLW storage facility on site and maintaining the safe enclosure.

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

A1.1.13 Fugen, Japan

The Fugen (called on Advanced Thermal Reactor (ATR)) is a proto-type heavy water moderated, boiling light water cooled, pressure tube type reactor with 165MWe, owned by the Japan Atomic Energy Agency (JAEA). 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. Therefore, the Fugen was shut down in 2003, after 25-year operation.

* 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.

The decommissioning programme of Fugen has been approved on Feb. 2008. Following the approval, the final stage of the Fugen was changed from stage two to stage three (decommissioning of the facilities). The programme consists of the following four periods scheduled with considering the transportation of remaining spent fuel, the accumulation of experience for the dismantling work and the decrease of radioactivity of highly activated materials; (1) Spent fuel transportation, (2) Periphery facilities dismantlement, (3) Reactor dismantlement and (4) Building demolition. It is expected that decommissioning will be completed by 2028. In the first period, spent fuel and heavy water are transported out of the Fugen reactor. It is also planned to dismantle some facilities contaminated with relatively low radioactivity. In the second period, equipments surrounding the reactor core will be removed allowing to access to the core and for installing the remote-controlled machine to dismantle the core. The reactor core will be dismantled in the third period, and empty buildings will be decontaminated, demolished and released from nuclear control.

The amount of the radioactive waste from the decommissioning was estimated with taking account of operation records on each facility. About 50 000 tonnes from the total 362 000 tonnes was categorized as low level radioactive waste, however it is expected to further reduce this radioactive waste to 10 000 tonnes by applying the decontamination processes.

As a part of the work in the spent fuel transportation period, the main steam system and the feeder water system etc. are now being dismantled in the turbine building. The remaining tritium in the heavy water system is also being removed by drying with aerating and/or vacuuming for facilitating the dismantlement of the heavy water system. Moreover, methods on dismantlement of the reactor core are being studied. Dismantling will be done under water cover for radiation shielding and dust suppression.

Unique attributes/challenges

- In Fugen, the heavy water was used as the moderator; as a result, relatively large amounts of tritium are remaining in the heavy water system. Therefore, it is required to prevent the inhalation of tritium by using tight protective gear in the dismantling work for the heavy water system, which results in degradation of workability. It was, therefore, planned to remove the tritium from the heavy water system as complete as possible before the dismantling work. Previously, tritium removal methods by through-air drying, vacuuming and heating were examined using both the equipments in the heavy water up-grade facility and the heat exchangers of heavy water pumps as test specimens. These processes were basically effective for removing the tritium.

New methodology developments

- Pyrolysis and "depressurized oxygen plasma" method have been developed for the treatment of used ion-exchange resin. The used resin is stabilized and decreased to one-twentieth in the volume with this method
- The Decommissioning Engineering Support System (DEXUS) has been developed for planning the optimal process of dismantling work and carrying out the dismantling work safely and precisely. The system consists of: the 3D CAD database of the entire plant of the Fugen, the "COSMARD" code to evaluate the workload, exposure of workers, waste arising and schedule of the dismantling, the "VR-dose" code for visualizing the dismantling process and estimating the exposure dose of workers in the radiation dose distribution, and the worksite visualization system to provide the necessary information to workers at the worksite using the AR technology.

A1.1.14 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 defueled under its operating licence and all fuel elements were shipped offsite for reprocessing by June 2001.

The decommissioning project was started in December 2001 and is expected to be carried out over 17 years in three phases. The site would then be in a “green field” status and will be re-used for the sitting of a new nuclear power plant. The first phase took 5 years, from 2001 to 2005 on schedule. 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 in the turbine building and the reactor service building were removed. The second phase began in 2006 and takes over 5 years. During the second five-year phase, the steam raising units and the primary gas ducts outside the reactor building will br 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 second phase the main activities are:

- Dismantling the SRUs and the primary gas ducts outside of the safe storage area.
- Removing the equipment on the operation floor of the reactor building and fuel handling building for the purpose of securing waste transportation routes.
- Installing radioactive waste treatment facilities at the fuel handing building (FHB) and cartridge cooling ponds (CCP).

  The SRUs are dismantled in turn from the bottom while lifting the SRU by large jacks to avoid spreading radioactive material. The remote cutting device is selected to obtain the knowhow on remote handling techniques to utilise reactor area dismantling in the third phase.

  And the following study concerning the reactor dismantling method started in 2007.

- Scenario, concept and procedure of reactor dismantling

The total costs for the Tokai 1 decommissioning project have been estimated at 885MUSD (681 MEuro) (2009). Over 60% of this total cost that is 538MUSD (414MEuro) is for radioactive waste management.

JPDR(Japan Power Demonstration Reactor) which was owned by former JAERI (Japan Atomic Energy Research Institute), ceased its operation in 1976 and it was utilized for the development and demonstration of decommissioning and dismantling techniques. Many of technologies obtained by decommissioning of JPDR are applied for Tokai 1 decommissioning. In addition to that, NUPEC (the National Power Engineering Corporation) has carried out various demonstration tests. The Tokai 1 decommissioning plan is established with input from these technologies.

Unique attributes/challenges

- Do It Yourself policy is introduced during the project by JPAC group. The practical know-how will be utilized in future NPP decommissioning in Japan.
- Waste disposal facilities are not constructed yet. LLW will be disposed of at a burial facility outside of the Tokai site. Very low revel waste will be disposed of at the Tokai site.

New methodology developments/new equipment or instrumentation developments

- Dismantling the Steam Riser Unit (SRU). The SRU is a large component (weight 750 tonne, height 25m, diameter 6m). A jacking-down method with remote dismantling device and 3D-CAD has been installed for this large component dismantling. The SRU is lifted by a large jack and dismantled from the bottom to top. This method eliminates work at high levels and minimizes the radiation control area.

- JAPC studied with a manufacturer to develop the remote dismantling device. A ring-shaped transfer device and an extraction system were developed for SRU dismantling. The remote dismantling device is operated semi automatically using a 3D-model and monitoring camera. A high power gas cutting device for thermal cutting and a disc cutter for mechanical cutting have been installed

Licensing and political issues

The Japanese Atomic Energy Commission’s statement “Framework for Nuclear Energy Policy” (October 2005) says that it is important to undertake the decommissioning under the responsibility of the installer. The installer, electrical power companies such as JAPC, did not have any experience with decommissioning. The JAPC struggled to carry on the project.

Tokai-1 is the first instance of a commercial nuclear power plant decommissioning in Japan. Decommissioning regulations and rules including safety regulation, waste disposal and cost estimating systems were not stable when the Tokai-1 project was commenced. The JAPC has encouraged to government to establish decommissioning system (regulation and institution) in Japan.

The Tokai-1 decommissioning project has encouraged the following issues:

- Cost funding systems to be established.
- Decommissioning safety regulation required amendment
- Regulation on waste treatment and disposal needed to be established
- Clearance system needed definition.

Lessons Learned

- Work efficiency is low in a confined space, especially in a non crane area at elevated ambient temperatures. This, combined with the lack of waste storage areas can impact cost and schedule and should be taken into account in the planning process.
- Existing utilities and operational infrastructure are utilized for the decommissioning project. This creates inefficiency due to conflict of priorities and interests.
- Documents necessary for project planning are difficult to gather.
- Some conventional components in the reactor building planned to be dismantled in the third phase have deteriorated and corroded making the job much harder - these components should have been removed earlier.
- Decommissioning cost is increased due to underestimation of waste volume. Accuracy enhancement of waste volume estimation is important for accurate cost estimation.

A1.1.15 Korea Research Reactors 1 & 2 (KRR-1 and 2)

The first research reactor in Korea, KRR-1, was a TRIGA MARK-II type (open pool and fixed core); its power was 100 KWt at its construction but was up-graded to 250 KWt by KAERI (Korea Atomic Energy Research Institute). This reactor played an important role for the development of the basic nuclear sciences and technologies and for fostering and developing nuclear industries in Korea. It had been operated until 1995 since the first criticality in 1962. The second reactor, KRR-2, was a TRIGA MARK-III type with an open pool and a movable core; its power was 2 MWt. Its first criticality was reached in 1972 and had been operated for 55 000 hours till the decision to decommission it in 1995. The main purpose of the KRR-2 was the production of radioisotopes and the application of reactor neutrons to such field as neutron radiography.

In 1996, it was concluded that KRR-1 and KRR-2 would be shut down and dismantled. A project was launched to decommission these reactors in January 1997 with the goal of a completion in

2008. The budget approved from the Government for the project was 20 million US dollars, including the cost for the waste disposal and for the development of ancillary technologies. The scope of the work during the decommissioning project was inclusive of dismantling of all the facilities, removing all radioactive materials from the reactor site and releasing the site and buildings for unrestricted uses. The operator of the decommissioning project was the KAERI because it was the licensee for their operation.

1997 – 2000: A plan for the decommissioning project of the KRR-1 and 2 was prepared. An immediate dismantling, minimum radioactive waste generation and development of necessary technologies were selected as the decommissioning strategies. A decommissioning plan including the EIA report was submitted to the Ministry of Education, Science and Technology (MEST) for the license in December 1998 and was approved in November 2000 after a safety review.

2001 – 2003: The dismantled facilities in this period were the auxiliary facilities of the KRR-2 which consisted of 12 laboratories, 10 lead hot cells and 2 concrete hot cells and which was used for the experiments with radioisotopes. In the hot laboratories all the apparatus and furniture was dismantled and removed and the ceiling, wall and floor were decontaminated. For the lead hot cells, all the facilities including lead bricks and concrete structures were dismantled and removed while after removal of highly radioactive sources, the inner wall was decontaminated and imbedded pipes were removed in the concrete hot cells.

2004: The core structure of KRR-2 were dismantled, cut into small pieces and packed into a shielded waste cask under water. The rotary specimens rack, inserted into the reactor core like a ring, was separated and moved to the pool of the KRR-1 to dismantle it with a developed tool. Besides the reactor core in the pool, there were many pipes and ducts, for irradiation of samples and a circulation of the water of the pool. The highly radioactive parts of the pipes were separated under water and the less active parts were pulled out of the water and cut into small pieces in a temporary shielding apparatus.

2005: Before the main cutting works for the shielding concrete, all the facilities embedded in the concrete, such as the thermal columns and the beam port tubes, were dismantled because they were highly radioactive. The cutting lines were designed with a help of the 3 dimensional radioactivity maps and the concrete was cut with diamond wire saws. After a removal of all the not-radioactive parts of the concrete, a green house with plastic sheets was installed to cover all the activated parts and a breaker was utilized to cut the remaining concrete into pieces, small enough to be packed into 4 m3 waste containers.

2006 – 2008: The laboratories, steel hot cells and lead hot cells of the KRR-1 were easily dismantled with experiences of the KRR-2 dismantlement. All the facilities in yard of KRR-1 and KRR-2, including underground liquid waste storage tank, hot cells, and waste treatment facilities were also dismantled.

2008 – 2009: After all the facilities were dismantled and removed except the core of KRR-1, the residual radioactivity has been measured and its impact to the environments and public members has been assessed by MARSSIM principles.

2010 – 2013: The core structure inside the shielding concrete of the KRR-1 was not dismantled because it was decided that it would be reserved for a monument. But in 2009, it was finally concluded that the core structure of the KRR-1 would be also dismantled due to the safety reason. The decommissioning plan will be reassessed in 2010 and the physical dismantling work will be started from 2011.

All the dismantled materials were classified into two groups; releasable and radioactive material. The radioactive waste which had higher radioactivity than a classification criterion (0.4 Bq/g) was packed into 200 liter drums or 4 m^3 containers. They are now temporary stored in the reactor hall of KRR-2 and will be transported to the national LILW repository when it is operational. The releasable waste was separately stored and managed. Among the releasable waste, about 1 800 tonnes of concrete waste was released and reused as base aggregates of a road bed after permission from the MEST. The metal waste will be reused in general steel industries and the combustible waste will be incinerated. For these, new facilities for incineration and metal meting are now under licensing.

Unique attributes and/or challenges

- There were no unique attributes and technical and political challenges for the projects.

New methodology development

- A matrix sampling on the surface and along to the depth of the shielding concrete of KRR-2 was carried out. From the measured radioactivity, mapping of the surface radioactivity was carried out and it was extended to 3 dimensional diagrams with a simulated dependency of the radioactivity along the depth. With the help of 3-D map, cutting lines, sequence of the cutting and cutting machines (diamond wire saw) were determined and detailed cutting procedures were established.
- A computer programme for the management of decommissioning information, named DECOMMIS was developed and used for the storage of the information and the data modification for the basic information for the project management. This programme gathered the data on man power consumption, project progress, cost, waste, radiation and so on. The DECOMMIS is now being improved for processing data for the next decommissioning projects.

New equipment

- The graphite blocks in the thermal column, especially those located near core of the KRR-2, were highly neutron-activated and had a very high radioactivity. In order to pull them out, a remote gripping tool was developed and utilized. The tool consisted of rubber mouse, which gripped the blocks by vacuum force, long arm and positioning mechanism. It was operated in a green house with temporary ventilation system.
- RSR dismantling machine. RSR was a radiation rack for samples and was inserted around the reactor core of KRR-2. The material of the RSR body was aluminium but the rotating mechanism, chain and bolt/nut, was made with stainless steel. The stainless steel was very radioactive and was decided to be separated to reduce the waste amount which should be packed in a shield cask. A machine was fabricated and successfully used for dismantling RSR under water. The equipment consisted of screw drivers, saws, fixing mechanism and air pressurized mechanical seals for leak prevention between moving and fixed parts.

Licensing and political Issues

- The release of the site and buildings for unrestricted use after decommissioning of KRR-1 and KRR-2 is the final goal of the project. The impacts by the residual radioactivity were assessed according to the MARSSIM principles. But no criteria for the release of the site after decommissioning were defined in the laws in Korea. The worldwide adopted criteria were ranged from 10 to 300 μSv/y. The results of the studies on these criteria, conducted by KINS (the Korea Institute of Nuclear Safety), set the desirable dose limit between 100 and 250 μSv/y. The permissible radioactivity remained in the KRR site was calculated on the base of dose limit of 100 μSv/y and all the contamination was removed to satisfy it. A final report was prepared and will be submitted for delicensing the site for the free release.

Lessons learned

- During the decommissioning project of KRR, the preservation of KRR-1 reactor core without any dismantlement for a monument was issued. Many discussions were held among the ministry (MEST), the owner of the site (KEPCO), decommissioning operator (KAERI), safety related institute (KINS) and the stakeholders who proposed and opposed the maintenance of the KRR-1 core for a monument. For a decision, a public opinion search was selected without any consideration on the safe restoration of the nuclear facility. The preservation of the KRR core was decided by 67 % affirmative side among the responded public. The dismantlement of the KRR-1 core was suspended and some works for the preservation including decontamination of the outer wall of concrete shielding and installation of a hard transparent cover over the reactor pool and a purification equipment of the pool water were carried out. In 2009, the pool water leaked out and flowed to outside of reactor building through corrosion holes of the aluminium lining plate of the pool. It was proven that there were no severe impacts to the environments by the pool water because the water was being continuously purified. But the plan for the monument was changed to the installation of a model instead of the preservation of the original core of KRR-1, as it were. The decommissioning plan will be reassessed in 2010 and the physical dismantling work will be started from 2011.

A1.1.16 Hamoaka NPP Units 1, 2

In Japan, terminated nuclear power stations must be dismantled and removed, and doing this requires establishing a nuclear reactor facility decommissioning plan as based on the Law on the Regulation of Nuclear Source Material, Nuclear Fuel Material and Reactors, and getting the approval of the national government.

Hamaoka Nuclear Power Station's Unit 1 and Unit 2 ended their operation on January 30, 2009. Their decommissioning procedure commenced after the submission of the "Application for the approval of the decommissioning plan for Hamaoka Nuclear Power Station Unit 1 and Unit 2" to the Minister of Economy, Trade and Industry on June 1, 2009 according to Article 43-3, Paragraph 2, Item 2 of the Law on the Regulation of Nuclear Source Material, Nuclear Fuel Material and Reactors, and subsequent approval for the application on November 18, 2009.

The application includes an overall plan for dismantling reactor facilities safely and surely, a description of tasks to be performed during the first period Chubu Electric is preparing to dismantle the facilities in the coming years (system decontamination, survey of facility contamination, etc.) and safety assurance measures, among other information.

Developments up to the end of operation at Unit 1 and Unit 2

Chubu Electric Power Company (CEPC)'s Hamaoka Nuclear Power Station is located in between Tokyo and Nagoya in Japan. The Power Station commenced the commercial operation of Unit 1 in March 1976 and Unit-2 in November 1978. Unit 1 and Unit 2 ended operation on January 30, 2009, and entered the decommissioning phase on November 18, 2009. There are two BRWs in decommissioning, two BWRs and one ABWR in operation and one new ABWR under planning for installation in the power station.

The replacement plan for Hamaoka Nuclear Power Station encompasses the decommissioning of Units 1 & 2 as well as the construction of Unit-6.

Unit 1 is a boiling-water reactor with the output of 540 000kWe. It launched commercial operation in March 1976. The reactor's operation was suspended in November 2001 following a piping rupture in the Residual Heat Removal system. The 19th annual outage started in April 2002, with a view to replace the core shroud and improve seismic margin.

Unit 2 is a boiling water reactor with the output of 840 000kWe. It launched commercial operation in November 1978. The 20th outage started in February 2004, with a view to replace the core shroud and improve seismic margin. However, from the economic perspective, CEPC announced the Hamaoka Nuclear Power Station Replacement Plan in December 2008 (decommissioning of Units 1 & 2 and construction of Unit 6), and ended the operation of Units 1 & 2 in January 2009. Unit 1 operated 27 years from the time of first criticality to the end of operation in November 2001. Unit 2 operated 26 years from the time of first criticality to the end of operation in February 2004.

Decommissioning plan for Unit 1 and Unit 2

As a measure associated with the closure of Unit 1 and Unit 2, CEPC filed a notice on December 22, 2008 that Hamaoka Nuclear Power Station's combined output is reduced from 4,884,000kW (Units 1 – 5 total) to 3,504,000kW (excluding the output from Units 1&2) according to Article 9, Paragraph1 of the Electricity Business Law.

CEPC also applied for the approval to adjust the Technical Specifications to stipulate that no fuel be loaded into Unit 1 and Unit 2, and was granted the approval on January 19, 2009.

In initiating decommissioning, licensees are required to file an application with the government for the approval of their decommissioning plan and gain government approval in accordance with the Law on the Regulation of Nuclear Source Material, Nuclear Fuel Material and Reactors addressing:

- Facilities subject to decommissioning and their dismantlement method.
- Management and transfer of nuclear fuel materials.
- Removal contamination caused by nuclear fuel materials.
- Disposal method of radioactive waste.
- Decommissioning schedule etc.

The decommissioning plan was compiled after the announcement of the Hamaoka Nuclear Power Station Replacement Plan. CEPC filed the application for the approval of the decommissioning plan with the government on June 1, 2009, partially amended the application on September 15, 2009, and gained approval for the plan from the Minister of Economy, Trade and Industry on November 18, 2009.

The decommissioning began on 18^{th} November 2009 and is to complete by the end of FY2036. The 28-year schedule is divided into four phases. The above application covered:

Overall Plan for safe and secure dismantlement of the nuclear reactor facilities

Description of work to be performed during the preparation stage (Phase 1) in the coming years (system decontamination, investigation into the contamination status of facilities, etc.), and associated safety measures

Basic decommissioning policies (Safety assurance measures)

- Maintaining and managing facilities required for safety assurance.
- Preventing the leakage and diffusion of radioactive substances.
- Protecting dismantlement personnel from radiation.

Dismantlement methods

Phase 1: Preparation Stage

Phase 1, which runs for approx. 6 years from FY2009 to the end of FY2014, prepares for dismantlement work by removing nuclear fuel, conducting an investigation into contamination status, performing system decontamination, and dismantling out-of-service facilities/equipment outside RCA.

Transporting and transferring nuclear fuel

There are 1,529 fuel bundles at Unit-1 and Unit-2 as of the end of December 2009.

There are 69 spent fuel bundles and 96 new fuel bundles at Unit-1 and 1 164 spent fuel bundles and 200 new fuel bundles at Unit-2. All the nuclear fuel is to be removed during Phase 1. Spent fuel will be transported to the fuel pools at Unit 4 and Unit 5 or a reprocessing plant, while new fuel will be sent to a fuel fabrication plant or other facilities. The 68 spent fuel bundles in the storage pool at Unit 1 (only used for short time cold critical testing) and 148 new fuel bundles in the fuel storage pool at Unit 2 will be transported to Unit-5 to determine how to use them.

Investigating and examining the contamination status:

The contamination status of facilities is to be investigated to determining the dismantlement timing for facilities/equipment within RCA, explore a dismantlement method, assess the volume of dismantlement waste, and work out the period of safety storage. The investigation calculates the level of radioactive concentration for buildings, equipment, piping, etc., attributable to radio-activated and secondary contamination, while also collecting samples to improve the precision of assessment.

System decontamination

System decontamination is performed on equipment and piping prior to their dismantlement to remove radioactive substances affixed to their internal surfaces, from the perspectives of reducing exposure of radiation work personnel and area residents, preventing the leakage of radioactive substances in or outside the facilities, and reducing the amount of waste. Decontamination covers the Recirculation system, Water purification system, Residual Heat Removal system and reactor vessel.

Dismantling and removing facilities and equipment outside RCA

The out-of-service facilities and equipment (e.g. transformers), installed outside RCA, are dismantled and removed.

Phase 2: Dismantling/Removal Stage for Reactor Zone Peripheral Facilities

Phase 2, which runs for approx. 8 years from FY2015 to the end of FY2022, dismantles and removes reactor zone's peripheral facilities, which have a relatively low level of radiation.

Dismantling and removing reactor zone's peripheral facilities

Reactor zone's peripheral facilities, which have a relatively low level of radiation, are dismantled and removed gradually. In the application for the approval of the decommissioning plan, "reactor zone" is defined as the reactor vessel and its surrounding items including radiation shields.

Peripheral facilities represent facilities inside the turbine building, facilities in the reactor cooling system at the reactor building, etc.

Installing a facility for processing dismantlement debris

A dismantlement waste processing facility is to be installed to prepare radioactive waste, to be generated in dismantlement and removal work, for underground disposal through cutting into sections and putting them into containers.

Phase 3: Dismantling/removal stage for the reactor zone

Phase 3, which runs for approximativily 7 years from FY2023 to the end of FY2029, dismantles and removes items with a relatively high level of radiation such as core support structures, reactor vessel, radiation shields surrounding the reactor vessel, and containment vessel.

Phase 4: Dismantling/removal stage for building structures

Phase 4, which runs approximatively 7 years from FY2030 to the end of FY2036, removes radioactive substances, remaining on building walls and other surfaces through chipping once contaminated facilities are removed. The ventilation system and radioactive substance disposal facilities are dismantled and removed while paying due considerations to preventing the spread of contamination. After the confirmation of contamination status, RCA designation is lifted and the buildings are demolished.

Disposal of items contaminated by nuclear fuel materials

Decommissioning work generates approximatively 480 000 tonnes of waste at Unit 1 and Unit 2 combined. The waste is to be rationally sorted and appropriately disposed of.

Non-radioactive waste and waste that does not need to be handled as radioactive waste:

Of waste to be generated in decommissioning work at Unit 1 and Unit 2, non-radioactive waste and waste that does not need to be handled as radioactive waste (waste subject to clearance) account for 442 000 tonnes and 25 000 tonnes respectively. This is approx. 467 000 tonnes in total or 97% of the overall waste. These types of waste are to be recycled as resources as much as possible, or appropriately disposed of as industrial waste.

Low-level radioactive waste:

Of waste to be generated in decommissioning work at Unit 1 and Unit 2, low-level radioactive waste accounts for approx. 17 000 tonnes or 3%. Low-level radioactive waste is sorted according to the types of radioactive substances contained or the level of radiation based on laws and regulations, and appropriately put to underground disposal based on the classifications. Specific methods for disposal, including the disposal site, will be decided before the start of dismantlement work on reactor zone's peripheral facilities, and reflected to the decommissioning plan for approval.

Waste with a relatively high level of radiation (L1 waste):

Of low-level radioactive waste, waste with a relatively high level of radiation (L1 waste) includes the top guide, shroud, core support plate, etc. in the reactor core, and comes to approx. 200 tonnes for Unit 1 and Unit 2 combined. This type of waste must be "disposed of underground at the depth of 50 meters or greater (sub-surface disposal)" according to relevant laws and regulations.

The Nuclear Safety Commission is currently examining the guidelines for safety screening.

Waste with a relatively low level of radiation (L2 waste):

Of low-level radioactive waste, waste with a relatively low level of radiation (L2 waste) includes the reactor pressure vessel and the steam separator in the pressure vessel, and comes to approx. 2 200 tonnes for Unit 1 and Unit 2 combined. This type of waste must be "disposed of in a similar way to the near-surface disposal, currently provided at the Low-Level Radioactive Waste Disposal Center (near-surface pit disposal)" according to relevant laws and regulations.

Waste with an extremely low level of radiation (L3 waste):

Of low-level radioactive waste, waste with an extremely low level of radiation (L3 waste) includes pumps/pipes in the Recirculation or Core Standby Cooling systems, pipes in the Main Steam or Make Up Water systems, and other metal and concrete debris, and comes to approx. 14 000 tonnes for Unit 1 and Unit 2 combined. This type of waste must be "put to near-surface disposal without solidification into cylinders or installation of artificial structures (near-surface trench disposal)" according to relevant laws and regulations.

Campaigns for building understanding

Understanding of local residents and government officials is essential in proceeding with the decommissioning work. The company website is used to start disseminating information. Other campaigns include sending direct mail, organising briefings for local residents, and operating the information caravan (for mini-briefings for the public).

Unique attributes/challenges

The decommissioning plan for Hamaoka Nuclear Power Station Unit 1 and Unit 2 represents Japan's first decommissioning of a commercial lightwater nuclear power plant.

The decommissioning will be implemented to replace the power generation from Unit No.1 and 2 with new Unit No.6.

Two commercial BWRs will be decommissioned co-existing with the operational power The decommissioning will be implemented maintaining seismic stability.

The priority is given to the safety to steadily implement the decommissioning with transparency, and to the acquisition of trust from everyone.

International cooperation is very important

A1.1.17 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, CO2 cooled, pressure tube reactor. Two accidents took place in the facility and in the second (in 1977), fuel overheated and there was leakage of fission products into the primary system and the moderator. A decision was taken in 1979 to shut down the plant and decommission it.

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

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

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

The high surface dose rate from the fuel cans.

The high volume activity in the spent fuel (long-term storage) pond.

The high levels of contamination on the upper parts of the cans.

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

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

The 20 000 kN super-compactor 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 80oC before being fed into the thin film evaporator, to which bitumen is fed at 110oC. 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 109 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 1 011 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 1 010 Bq/l. To be treated are solutions up to 1 011 and 1 012 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 inhibitor 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.

Phase 1 (radiological safe status) has essentially been achieved and focus has now shifted to the shut down of the V1 Npp's.

A1.1.18 Bohunice V1, Slovakia

By government decision, units 1 and 2 of Bohunice V1 NPP will be shut down in 2006 and 2008 respectively. The reactors will be in a shutdown operations phase until 2012 under the responsibility of JAVYS Nuclear and Decommissioning Company. Dismantling activities are anticipated to continue through 2025 at which time the site will be released.

During the shutdown Phase, the work has been subdivided into three groups:

- Group A – physical shutdown and cleanup operations.
- Group B – decommissioning documentation.
- Group C – RAW treatment. The V1 NPP project can take advantage of RAW treatment facilities already available through A1 NPP decommissioning programme.
- Group D Projects – EC-combined programme documents (financial perspective).

A review of Group A Projects achievements in this reporting period

- Reconstruction of Area Protection systems completed.
- Reconstruction of the Public Warning and Notification System completed.

- Relocation of the Emergency Response Centre completed.

A review of Group B Projects achievements in this reporting period.

- V11 NPP Decommissioning 1st Stage Plan & completed.
- A Decommissioning database has been established.

A review of Group C Projects achievements in this reporting period.

- Treatment and conditioning of historical waste has begun.
- Interim storage of RAW at Bohunice site established.

A1.1.19 JEN-1, PIMIC project, Spain

Under the responsibility of the owner of the research centre (CIEMAT), ENRESA (Main Contractor), carried out from 2006 to 2010 the dismantling of several facilities and installations, including an experimental nuclear reactor that was phased out a few years ago. The Regulatory body approved the Decommissioning Plan (2005) and the Environmental Ministry approved the Environmental Impact Assessment (2005). The municipal licence was issued by the end of February 2006. The global authorisation was issued by the end of 2006.

Relationships between ENRESA and CIEMAT were initially defined in the Director Plan under the supervision of Regulatory Body. During the project ENRESA and CIEMAT have established procedures of cooperation in several subjects: radiological protection, industrial safety, interactions with regulator, etc.

The general objective is D&D of redundant facilities, upgrade of other buildings and facilities, restoration or affected areas and land, and to generally improve the safety culture. The project includes the establishment of several auxiliary installations:

- 2 RadWaste Storage Areas – Building 55 & Building 11.
- 1 Waste clearance centre – Building 63.
- 1 Waste characterisation & conditioning centre - Building 64.
- 1 Waste centre.

The D&D activities were broken down into subprojects, mainly in coincidence with the two principal areas, East & West:

East Zone

- Reprocessing plant – Building 18.
- Liquid radioactive waste conditioning plant – Building 13.
- Liquid radioactive waste storage area – Building 53.
- Lentil zone.

West Zone

- Reactor building and associated systems.
- Underground tanks.

Other

- Montecillo Area.

In these zones were carried out the main tasks included in the decommissioning project such as: dismantling, decontamination, radwaste conditioning, release, demolition, restoration and others (see the percentage in progress, Table 1.)

The dismantling project is scheduled to be complete by the end of 2010 and cost a total of 22.5 M€ from 2005.

At the present time release methodologies for materials and surfaces are licensed and are being applied.

The total official collective dose accumulated in 2009 was 5,8 mSv-person.

The total released materials expended from the beginning of the project untill 2009 were as follows:

Metallic Scrap :	251,41 tonnes
Isolated Wool):	1,9 tonnes
Rubble:	270,26 tonnes
Others:	14,49 tonnes
Total	538,06 tonnes

The total Radioactive Wastes, by categories, sent to the El Cabril facility were as follows:

Very Low Level Waste (VLLW):	116,8 tonnes ~ 101,1 m^3
Intermediate and Low Level Waste (ILLW):	412,4 tonnes ~ 396,5 m^3

The remaining activities for the 2010 & 2011 are as follows:

- Refill with recycled rubble the Underground Tanks area.
- Decontaminate the soil around Buildings 13 and 18.
- Lentil Zone:
  - Decontamination
  - Rad Waste conditioning
  - Refill

Lentil Zone & Montecillo Zone Restoration including Radioactive Waste (mining tails) management (3.000 tonnes) are expected to be started during this year.

Some of the unique attributes to this project include:

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

A1.1.20 Jose Cabrera Project, Spain

Stage A - Prior to the definitive shutdown of José Cabrera NPP

The Jose Cabrera NPP (JC-NPP) was the first power reactor developed in Spain and provided the base for future development and training. The reactor construction started in 1964 and it was officially on-line by 1969. The reactor was operational until 2006, at which time it was shut down and went into a transitional stage. The reactor is a Westinghouse 1-Loop PWR with a thermal power of 510 MW and net electrical output of 160 MW. The fuel was UO2 enriched with 3.6% U-235. The containment is reinforced concrete with a stainless steel head. Spent fuel was stored in a storage pool.

Previously, in 2003, ENRESA drew up the Basic Strategic Study for the dismantling of the JC-NPP, which was submitted to the Ministry of Economy.

The Basic Strategic Study proposed that all stages necessary for the complete dismantling of the plant be addressed immediately. Finally the restoration and release of the land on the site will be performed. ENRESA has chosen the immediate total dismantling IAEA Level 3 (Green Field) strategy.

Stage B - Transition stage

The transition stage in the dismantling of the JC-NPP was the period from definitive shutdown to granting of the dismantling permit and transfer of the ownership to ENRESA. The main purpose of this stage has been to reduce the potential risk of the installation. This consisted fundamentally of removing the spent fuel from the reactor spent fuel pool and transferring it to the independent temporary storage (ITS) facility and the conditioning of the operations radwaste. In addition, and with a view to prepare the site for dismantling, this stage has also included decontamination of the primary loop and characterisation of the installation.

On April 30th 2008, ENRESA issued a request for the transfer of the ownership of the JC-NPP to the Ministry of Industry, Tourism and Trade, including all corresponding authorisation documentation for the Dismantling and Decommissioning Plan as required by regulation. The transfer of responsibility to ENRESA took place in 2010 (11th February), with the project to be completed in six years.

Stage C - Transfer of the ownership

The specific nature of the Spanish case, implies the need for the ownership of the facility to be transferred from the previous licensee to ENRESA at the moment of granting of the dismantling permit.

The previous licensee and ENRESA have been working for more than three years on fine tuning a transfer agreement regulating and protecting the rights of both organisation assigning ownership and the assignee, as well as service performance contract under which a part of the operating personnel, some 50 persons, will continue to work under the ownership of ENRESA, collaborating in the dismantling tasks and contributing with their experience of the NPP, to improve the overall performance.

Stage D - Dismantling activities

The D&D project presented gives priority on safety taking into account environmental, radiological, the public and workers aspects. The main activities of the project have been broken down into five major sequential groups, which may in turn divided into several sub-activities with different relevant milestones in accordance with the associated plans that make up the project.

Phase 0 - Removal of fuel and preliminary works

- Post shutdown and primary system decontamination
- Characterisation update
- Inventory update
- Shutdown, drain and isolation of non required systems
- Construction of interim storage facility (ISFS)
- Transfer fuel to dry storage cask

Phase 1 - Preparatory activities for D&D

This phase includes the dismantling activities related with the site buildings and facilities without radiological impact. Among the most significant buildings and facilities is the Turbine building, Diesel building and cooling towers area. Other activities are as follows

- Mobilization and set up offices
- Hazardous material removal
- Dismantling of non-radioactive systems
- Turbine hall modification and adaptation
- Test and start up procedures for new systems

Phase 2 - Dismantling of major components

This phase will be carried out in accordance with the Radioactive Elements Disassembly Plan. It will be the most significant activity from the point of radiological aspects, cost and time and therefore requires the highest levels of safety using the most qualified and specialized companies and workers. The basic activities will be as follows:

Disassembly & Decontamination of the:

- reactor building
- auxiliary building
- evaporator building
- temporary waste storage facilities

Including in this phase will be the most complicated activities regarding with the segmentation of the major components of the NSSS, which are located in the Reactor building. The sequence of the disassembly proposed activities will be as follows:

- reactor internals
- reactor pressure vessel
- major components: main coolant pump, steam generator & pressurizer

Phase 3 - Removal of auxiliary installations, decontamination and demolition

The Surfaces and Structures Decontamination Plan, the Activated Concrete Plan and the demolitions and backfill Plan will be applicable to this phase. The decontamination will be performed after removal of the components from the different buildings.

- Equipment removal, decontamination, clearance survey and demolition

Phase 4 - Environmental restoration

The Site restoration Plan will serve to guarantee that the land and buildings will be free of any residual radioactivity. Consequently, this phase will include the application of the different release methodologies, the demolition and the refilling of certain cavities or areas.

- Final cleanup and site restoration, close-out report, de-licensing and return

Greenfield is the final end state objective. Like radiological characterisation, materials management is an activity that is performed throughout all the phases of the project. The dismantling will generate a large quantity of waste materials, and it will be necessary to determine which should be recycled or managed as waste. Clean concrete rubble may be reused on site, conventional metallic materials will be recycled and toxic & hazardous products will be deposited and treated in appropriate facilities via authorized management organisations.

More than 104 000 tonnes of materials will be generated during D&D activities. These quantities are divided as follows:

- Spent Fuel 175 tonnes
- ILW 43 tonnes
- L&ILW 2 492 tonnes
- VLLW 1 443 tonnes
- Conventional Waste 100 909 tonnes

The cost assessment carried out at the beginning of the project (2003) continues to be valid, the estimated budget amounting of 135 million Euros (including a 20% contingency). This figure does not include close to 35 million Euros for the management of the spent fuel which has constituted an independent project.

Phase 0 of the project was completed prior to turnover to Enresa. Phase 1; preparatory activities, are currently ongoing. This includes the disconnection of electrical systems and the removal of trench cables.

A1.1.21 Barsebäck NPP

Barsebäck NPP, two BWR of Westinghouse design – 1 800 MWt, 615 MWe, are located in southern Sweden on the west coast of Skane.

Barsebäck 1 was shut down in Nov 1999 and Barsebäck 2 was shut down in May 2005 according to political decision.

Barsebäck Kraft AB (BKAB) is the licensee owner and has the responsibility for the decommissioning of the site under a project called Barsebäck NPP-Decommissioning.

Barsebäck NPP is the first NPP in Sweden under decommissioning.

Complete documentation of this project will be important and lessons learned will be applied to decommissioning the rest of the Swedish NPP: s.

The basic strategy is to bring both reactors to a safe radiological state in which they can be efficiently and economically managed until a final repository for dismantling waste is available. This is now anticipated to be 2020 at the earliest. Pre-project activity will resume approximately 5 years before the availability of the repository.

BKAB's approach for the dismantling is:

- Safer – Eliminate and reduce risk by a systematic decontamination, start the dismantling by taking out large components, well-prepared steps, a Safety Analyze Report for dismantling.
- Faster – Create conditions for a flexible logistics by detailed characterisation, requirements for the dismantling in place, a good planning process, the process for each type of waste clarified in detail, the final disposal for dismantling waste in operation.
- Cost effective – Focus on the time-schedule by planning for less than five years time for dismantling of the site. Focus on the end point.

The ultimate aim of the decommissioning project of Barsebäck NPP is that the remaining buildings of the plant including equipment should be declared free-released for re-use or dismantled, whichever is appropriate at the time.

Barsebäck 1 and 2 have been in Service operation (Care and maintenance) since December 2006, when all fuel has been transported away to interim storage at CLAB in Oskarshamn.

The Organisation at BKAB has gone down from 450 during operation of both units to 50 employees involved in Service operation. There was a re-organisation at BKAB in 1 January 2007 and the company is organized in the following areas of function:

- Site service operation
- Planning for the dismantling
- New business

According to the plans for dismantling of Barsebäck NPP it will take 4 years of projecting, 5 years of free release of the site and 2 years to restore the site depending of chosen End state.

The main activities that have been completed or are ongoing for the Service Operation (Care and Maintenance period) include:

- An overall decommissioning plan has been presented and accepted by the owner and the Authorities.
- All nuclear fuel has been sent to interim storage at CLAB in Oskarshamn.
- Hazardous materials such as turbine oil and chemicals have been removed from site.
- Some preventive maintenance has been switched to corrective maintenance.
- Inventory of existing documents has been done and continues.

- BKAB have built up contact nets and competence by taking part in different kind of national (SKB) and international missions (IAEA, OECD/NEA, WNA, ENISS, WANO, E.ON, Vattenfall Europe, EPRI)
- A new Management system, a new Safety Analyze Report (SAR) and a new Safety Technical Regulation (STF) for Service operation has been created and sent to the Swedish Authorities.
- Project ANPEL. Rebuilding of the electricity systems and operation systems. The goal was to adjust the electrical systems for the actual demands and requirements for the Service operation and to create a site easier to survey and to reduce costs for operation and maintenance.
- Project Decontamination. Decontamination has been done, during 2007/2008, at Barsebäck 1 and 2 of the primary systems and the lower parts of the reactor tank with an excellent result.
- The Central control room is unattended since 17 December 2007 and the supervision of the Service operation is handled by a system of duty engineers (VDI) and alarm operator (LOP).
- Energy saving activities has been performed via using lesser electrical lighting, reducing heating in buildings and optimization of the ventilation system.
- A project has analyzed existing Swedish laws, ordinance and regulations to estimate the affect on future plans for dismantling of Barsebäck NPP.
- Two studies have been done to compare the possibility to take out the whole reactor vessel in one piece or to do segmentation. BKAB recommend to SKB that we during the future dismantling deliver the vessel in one piece to SKB for transportation and final disposal.
- BKAB have started up projecting for the site characterisation of contaminated material, buildings and soil including hazardous materials.
- Developing a plan for the operational waste in the RH pools (internals from the operation phase) has started up.
- A study called Large components for the future demolition and disposal of components, such as the turbine, condenser, heathers etc, has started up.
- BKAB received a temporary permit, to stay in Service operation, by The Environmental Court during 2006. The Permit goes out at 31 December 2012. An ongoing working party is writing a new Environmental Impact Assessment just now.
- Activities, under the department New Business, are using the closed down facilities for external purpose such as Barsebäck Test and Maintenance Centre with different kinds of courses for personals from the other Nuclear Facilities in Sweden. The Plants are also used for external testing of equipment and construction solutions for the other NPP's in Sweden. The Department also sell out components not needed during Service operation and during dismantling.

Unique attributes/challenges

- Barsebäck NPP are the first Nuclear Power Plant that goes into decommissioning in Sweden and our experiences can be of a important guidance for other NPP's.
- Barsebäck closed down because of political reasons and have reduced the Staff from about 450 during operation of both units to 50 during Service operation without dismiss anyone.
- It is a challenge to be in Service operation for about 15 to 20 years and our Lesson learned can be of use to other NPP's round the world.

Lessons learned

Below brings out some important lessons learned from Barsebäck NPP decommissioning project:

- The re-organisation during 2000 when Unit 1and its employees focused on Decommissioning and Unit 2 focused on operation is important to maintain safety and cost optimization on the site.
- To have a social and economical programme for the employees is important.
- An open dialog with stakeholders, as an example the Authorities creates confidence.
- The optimal way to prove and to make influence on new regulations from the National authorities is to start that process early on the international arena, IAEA and EC.
- A good planning process is most important. If you have a minor organisation it is very easy to underestimate the needs from your own personal with vital knowledge from the sites construction. Barsebäck found out that some key electrical personnel were needed in some projects ongoing at the same time. Do a good analysis of the projects and do not miss to compare them to find out if there are any overlaps of key personals.
- To create good working conditions on the site it is beneficial to do system decontamination as early as possible.
- Before staff with experience and knowledge of historical events leaves the site or company try and record this knowledge.
- Updated documentation is vital during the whole decommissioning process. There can be a lot of useful information in private archives.
- Mind setting from operation to decommissioning is a key to success. BKAB's experience is that an operation based organisation needs external helps from international experts for decommissioning expertise.
- Clear the site from hazardous material as soon as possible. It is vital for the Safety Analysis Reports that have been done for the Service operation.

A1.1.22 Studsvik R&D Reactors, Sweden

The announcement of the shutdown of the Studsvik Reactors was made in December 2004 and actual shutdown was realized in June 2005. The reactors were primarily research reactors used for materials testing, fuel development, neutron radiography, medical applications and a variety of other uses. The reasons for shutdown were mainly economic based. Studsvik Nuclear AB holds the operating license and is responsible for decommissioning the reactors but in the spring of 2009 it was decided between Studsvik AB and the owners of the Swedish nuclear power plants that the responsibility for the reactor decommissioning project shall be transmitted from Studsvik Nuclear AB to the company AB SVAFO, owned by the Swedish nuclear power industry, at the latest 2011-12-31. There will be no changes in the lay-out of the project due to changes in responsibility.

AB SVAFO is the company that earlier had been responsible for decommissioning of the Active Central Laboratory (ACL) at Studsvik site. The ACL decommissioning project was a CPD/TAG project and the laboratory building was demolished in 2006.

Two strategies were considered for decommissioning the R2 reactor facility:

- Prompt decommissioning (complete 2017).
- Deferred decommissioning (complete 2032).

Prompt decommissioning was chosen enabling the efficient use of existing experienced staff and infrastructure support.

The decommissioning project is divided into four phases:

- Shutdown – removing fuel from the reactor facility.

- Decommissioning – removing equipment from the reactor basin, dismantling the reactors and the reactor systems, emptying the basin, decontaminating and covering of the basin.
- A period of continuous decommissioning until characterisation and free release is achieved.
- Demolition of the buildings.

Background

The reactors were licensed in ten year periods. The reactor facility was upgraded during the license period of 1994-2004 due to requirements from the Swedish Radiation Protection Authority and a new company strategy for the reactors, more related to medical use. The reactor, instrumentation and safety systems were upgraded or replaced with new ones, safety documentations were revised and the documentation was upgraded. A Boron-Neutron-Capture-Theraphy (BNCT) facility were built in the late 1990´s for cancer treatment of brain tumors, glioblastom multiforme and would further on have been used for other kind of cancer treatments.

In the spring of 2004 the Swedish Radiation Protection Authority revised the safety status in the facility and the programme for upgrading the reactors and the facility and responded with satisfaction. The reactors were licensed in 2004 for another ten years of operation until 2014 with the option of another ten years period of operation, until 2024.

Studsvik was introduced to the stock market and became commercial with a couple of private main share holders as the new owners of Studsvik. The Board of Directors explained in December 2004 that there was relatively poor profitability regarding the reactors and informed that the Studsviks Board of Directors, because of that, decided that the reactors have to be shut down. Studsviks main focus after final shut down of the reactors has been incineration and melting of waste and scrap in co-operation in decommissioning projects world wide.

After the decision to shut down the reactors, projects were started to secure reactor operation until end of June 2005, to plan for decommissioning and to prepare and implement staff layoffs. The goals for reactor operation was to fulfil official requirements, secure important functions in the organisation and to fulfil irradiation and production requirements such as planned power transient tests (ramp tests), neutron transmutation doping of silicon, normal production of medical and industrial isotopes and to carry out planned BNCT treatments.

The reactor organisation fulfilled all the requirements imposed by the company management. There was a very high effectiveness in experiments, tests and production and there was a sound financial result at the reactor facility and the isotope centre during the spring of 2005.

The reactors were shut down finally on 16 June 2005. The shutdown of the reactors also meant the end of the activities at the isotope centre, the neutron research centre and the cancer treatment facility.

Summary of the final six months of operation:

- The organisation fulfilled all the requirements imposed by the authorities and the company management.
- Careful planning gave fulfilment of objectives.
- Very high effectiveness in production.
- Sound financial results at the reactor facility and at the isotope centre.
- Good climate of cooperation in the facilities during the entire process.

Decommissioning actions

2005, first half year

Plan for and implement decommissioning such as; winding up contracts with the customers, prepare for defueling, removing equipment and fixture sand general decommissioning planning.

Two alternative decommissioning strategies were prepared:

1) Shutdown, complete decommissioning followed by demolition of the buildings, completed by turn of year 2016/2017.
2) Shutdown followed by some decommissioning steps and then service operation, re-establishment after about fifteen years, continued decommissioning, then demolition of the buildings, completed in the turn of 2031/2032.

Decommissioning was divided into four phases:

1) Shutdown characterized by cutting fuel and removing fuel and test rods from the reactor facility.
2) Decommissioning divided into two stages where the main task in stage 1 involves removing loose and fixed equipment from the reactor basin, dismantling the reactors and reactor systems, emptying the basin, decontaminating the basin surfaces and then covering the basin.
3) Service operation (possibly) with the facility put in "mothballs" with general supporting systems, radiation monitoring, active drainage and ventilation systems in operation.
4) Demolition phase after free release on the facility and approval of the supervisory authority.

The original emphasis was on continuous decommissioning with the option to change to the alternative with interim service operation.

2005, second half year-2006

- Day-to-day operation of the facility.
- Cutting and transportation of fuel to the fuel storage building.
- Work with "loose" experiment and irradiation equipment in the reactor basin.
- Dealing with equipment, scrap and waste in the buildings.
- Dismantling of active cells.
- Revising the control programme for reactor systems.
- Implementing control and maintenance programmes.
- Radiological survey – stage 1.
- Application to the Environmental Court based on environmental impact and technical description documentation.
- Inventory of spare parts, equipment, scrap and waste.
- Decontamination.
- Documentation.
- Putting a decommissioning archive in order.
- Archiving.

Shutdown operation was completed in 2006 when all fuel had been removed from the reactor building.

2007-2011

The main activity planning for the period is to "secure" the reactor facility by means of dismantling the reactors and associated systems in the reactor basin, emptying out the water, decontaminating the basin liner and then covering the basin. By dismantling the reactor systems with high induced activity a secured radiological state can be reached in the reactor facility. After that the control requirements are less stringent and the remaining reactor systems in operation can be shut down.

The licensing procedure for dismantling are divided into two phases:

Phase 1	Obtain a Permit from the Swedish Environmental Court.
	The Environmental Court judgment that was handed down in March 2007 specified continuous decommissioning with subsequent demolition of the buildings.
Phase 2	Obtain a Permit from the Swedish Radiation Protection Authority
	The Swedish Authority has the responsibility to inform the European Commission about the planned dismantling activities. The Authority informed the European Commission by sending the article 37 report to the Commission in January 2009.
	The European Commission responded in March 2009 with complementary demands before the Commission will consider the report.
	A final article 37 report is planned to be sent to the EU-Commission before the summer of 2010.
	Before the Swedish Radiation Protection Authority is allowed to give permission for dismantling of the reactors and the connected systems they have to await the Commissions approval of the report.
	Dismantling of the reactors and the connecting systems in the basin was planned to start in 2009. Due to the Swedish Radiation Protection Authorities delay in account to the EU-Commission of the article 37 report, the dismantling is re-planned to start in the beginning of 2011. The project plan had to be revised because of this.
	Survey and decontamination of the experimental loop machine rooms was done in 2008. Focus after that has been on preparing activities for dismantling of the reactors, the reactor systems in the basin, the experimental in-pile loops and the connected machine rooms.
	All reference documentation from reactor tank replacement in 1984-1985 have been brought up again and evaluated regarding the dismantling procedures.
	Dismantling of the reactors has been divided into steps and are planned in instruction packages.
	The instruction packages are detailed descriptions for the dismantling operations. The packages are adapted to guarantee both radiological and working safety. After the instruction packages have been settled, separate time schedules have been established for each instruction package. A strategy for measuring dose rates in the reactor environments and at the connected systems have been settled and basic measures of dose rates have been carried through. The instruction packages together with the dose rates and the time schedule for performance have given the preliminary dose budgets for dismantling in stage 1. A characterisation plan for all categories of scrap and waste, for stage 1 of dismantling, and descriptions for all categories of scrap- and waste packages and containers have been settled. Waste handling instructions for all categories of scrap and waste have also been settled regarding; sorting, handling, treatment, packing and transportation. Analysis of radiological and working environmental risks has been carried out and radiation protection instructions have been adapted to the coming situation with dismantling.
	There have been continued radiological survey in the facility as a follow up to the basic survey carried through 2006 and the main radiological survey are planned to start in the summer of 2010.
	During the period the facility is also to be emptied as far as possible as regards loose equipment, loose materials, scrap and waste.
	Updates of the decommissioning plan, safety reports and safety operation assumptions have also been worked out.
	Structuring and filing documentation of the facilities and systems, the building and system drawings and the technical notes of the system functions and construction have and are to be taking care of.

2007-2009

- Transport of MTR fuel to DOE.
- Documentation, preparation of documentation required by the authorities.
- Application to the Swedish Nuclear Power Inspectorate for (partial) dismantling of the reactors and reactor systems in the reactor basin.
- Planning of dismantling stages.
- Preparation of work sequence orders.
- Drawing up dose budgets.
- Preparations for heavy lifts.
- Preparations of transport containers.
- Plan packaging and transportation.
- Day-to-day operation of the facility.
- Implementing control and maintenance programmes.
- Radiological survey – stage 2.
- Work with “loose” experiment and irradiation equipment in the reactor basin.
- Dealing with equipment, scrap and waste in the buildings.

2012-2016, first half year

In stage 2 of decommissioning external contractors will be engaged for dismantling of fixed reactor systems in the buildings and for radiological survey and for decontamination.

The present decommissioning organisation staff will function as supervisors for the contractors.

2016, second half year-2017, first half year

Conventional demolishing of the buildings, planned to the second half of 2016, after the facility have been free released by the authority. In the spring of 2017 there will be work with the final documentation.

Unique attributes/challenges

Due to the different activities carried through when the reactors were in operation such as a large range of experimental and irradiation activities there has been a historical investigation about occurrences that may have an impact on the decommissioning.

New methodology developments

New waste handling instructions, regarding dismantling and decommissioning, for all categories of scrap and waste types have been taken out.

New equipment or instrumentation developments

Dismantling of the reactors and connected systems in the basin, decommissioning stage 1, will be done with special manufactured tools and the cutting of reactor parts will be done in a separate pool in the basin.

Licensing and political issues

Due to the Swedish Radiation Protection Authorities delay in account to the EU-Commission of the article 37 report, dismantling of the reactors was re-planned to start in the beginning of 2011, instead of 2009. The project plan was revised but the total time schedule is the same as the original one, because there have been other steps in decommissioning that have been carried out, earlier than planned.

Lessons learned from decision to final shutdown

After the decision to shut down the reactors finally there was requirements imposed by the company management to secure reactor operation for another half year.

The lessons learned during this period were:

- The decision to continue reactor operation when informing staff of closedown gave stability in the organisation.
- Strong loyalty among the staff was noticed.
- Essential to have frequent information from the management concerning the ongoing process.
- Internal reviews important.
- Frequent presence of the management directly in the organisation was important.
- The time for shutting down reactor operations in connection with informing staff of layoffs was perceived as far too long.
- Continued operation with staff that had been given notice was not a problem.

A1.1.23 Taiwan Research Reactor (TRR), Chinese Taipei

The Taiwan Research Reactor (TRR) at Lung-Tan, Tao-Yuan, (about 50 km south of Taipei) is a 40 MWth, natural uranium, heavy water moderated and light water cooled reactor that operated between 1973 and 1988.

The reactor had been defueled and the heavy water removed by 1990. There was an evaluation for the feasibility of partially dismantling of the TRR, and to build a light water moderated, pool type, multi-purpose research reactor at the same site. During the same period, after radiological characterisation, all systems and components within 5m of the reactor block, including heavy water system, cooling system, neutron experiment facilities etc., had been dismantled and removed before 1994. The remodel project was terminated in 2002 and complete decommission for TRR launched. The decommissioning project of TRR was officially approved in 2004. In which the final goal is to dismantle all radioactive contaminated equipment, structures, and material in TRR within twenty-five (25) years. The budget for TRR Decommissioning Project was estimated to be 900 million NT$ (approximately 30 MUS$).

The reactor vessel of TRR was separated and lifted from its base and removed in one-piece to a specially built building for safe storage. It will be segmented before 2028 to comply with regulations. Besides basic monitoring on the reactor vessel, INER has begun to collect and collate the technical information needed for dismantling of TRR reactor internal components. The reactor hall and some utility systems were reused as metal decontamination site.

Major decommissioning activities at present stage are the contaminated spent fuel pool clean-up and spent fuel interim storage.

Unique attributes/challenges

- The TRR wet storage block was a 8m(L)x8m(W)x9m(H) concrete structure used to store fuel temporarily for fuel inspection and fuel loading operations. After the major earthquake in 2000, it was found that several pre-found cracks were growing significantly. The wet storage block was dismantled in 2004 with basically wire saw cutting method.
- The regulation for release of radioactive material in Taiwan was enacted in 2004. Over ninety percent of the concrete waste of the TRR wet storage block was crashed into gravels and delivered as recycled concrete and achieved the milestone for the first case of radioactive waste release and reuse in Taiwan.

- The TRR spent fuel pool is a reinforced concrete structure connected to the reactor building. After TRR's shutdown, the pool was used to its extreme to temporarily store highly activated material produced from implementation of TRR deactivation. Furthermore, in early 1990s, fuel particles from severely broken fuel was discovered spread over the whole spent fuel pool. Particles over 20µm have been retrieved in filter cans. Further attempts on retrieval of smaller fuel particles were not successful. By mass balance, there are still 140 Kg of uranium (powder) in the spent fuel pool. Major waste inventory includes 2 500 aluminum flow tubes, sludge and debris of oxidized uranium powder containing 140 Kg of uranium, 35 canisters containing metallic uranium spent fuel element, 4 test rods, filters, baskets and some other miscellaneous items etc. The clean-up of the highly contaminated, without liner, concrete structure pool makes the major challenge for TRR decommissioning project at this stage.

New methodology developments

- There were 2 500 flow tubes sections stored in the spent fuel pool. These tube sections are activated, and the maximum dose rate is around 20mSv/h. Oxidized uranium powder/sludge accumulates on their corroded surfaces during underwater storage for more than 20 years. Ultrasound bath cleaning method was used to clean-up those tubes so that they can be categorized as low level waste instead of greater than class C (GTCC) waste.
- There are 39 fuel canisters stored in the spent fuel pool. These canisters contain deformed fuel rods, damaged fuel rods, intact test rods, and filter cartridges containing oxidized uranium powder. The uranium sums up to about 1 500 Kg. To clean up the spent fuel pool, these spent fuel canisters have to be moved out for interim storage safely. Nevertheless, uranium metal tends to react with oxygen when in contact with air or moisture. Since there is only relative small amount of metallic uranium spent fuel and the risk of chemical reaction during storage must be eliminated, INER decided to convert the spent fuel from metallic form to stable oxidized form, such as U3O8, before the interim storage. The process developed for spent fuel stabilization includes fuel segmenting, fuel de-cladding, uranium metal oxidation, oxidized product sealing, and oxidized product dry storage.
- For the amounts of Pu and U contained in the spent fuel, a non-destructive material accounting system (PCC, plutonium coincident counter) was developed and used in the oxidization process.
- To collect the uranium powder deposited at the bottom of the pool, INER takes the precipitation method instead of filtration method for the preliminary powder collection. The collected sludge with uranium powder is planned to go through the oxidation process and to be under dry storage with oxidized product of TRR spent fuel.

Lessons learned

- TRR spent fuels are stored in the pool for over 20 years, moisture has intruded into the deteriorated canister. A risk of direct dry storage of TRR spent fuel is occurs from the uncontrolled oxidation reaction of UH3 spread in the uranium metal lattice. In the oxidization process, safety measures are necessary. Argon spray is used to restrain the unbridled oxidization condition in the hot-cell process.

A1.1.24 Windscale advanced gas-cooled reactor (WAGR), United Kingdom

History

The WAGR was a 33MW (E) prototype reactor built to study the performance of gas cooled fuel elements suitable for a commercial reactor, to serve as a test bed for further development of advanced fuel and other components and to provide operational experience of power production. WAGR was the forerunner of a family of 14 reactors on 7 sites in the UK.

WAGR was constructed between 1957 and 1961 and was a carbon dioxide cooled, graphite moderated reactor using uranium oxide fuel in stainless steel cans. The graphite moderator 15ft (4.6m) diameter and 14ft (4.3m) high was housed in a cylindrical reactor vessel with hemispherical ends enclosed in a domed steel containment building 134ft (40.8m) high and 135ft (41.1m) maximum diameter.

In anticipation of the UK's likely nuclear decommissioning needs the UKAEA decided in 1981 to decommission WAGR to the International Atomic Energy Agency's (IAEA) stage 3 as the national demonstration exercise for power reactor decommissioning. In 1993 following a project review, the project was limited to removal of the reactor core and pressure vessel with the concrete bioshield and iconic containment building remaining until 2045. In 2005, to address the Nuclear Decommissioning Authority's vision to accelerate decommissioning, plans were laid to completely demolish the WAGR complex to a brownfield site suitable for restricted reuse by 2015. However in 2008 the amalgamation of Windscale with the greater Sellafield site caused WAGR funding to be reallocated to higher hazard projects and the deferral period is again in the plan.

The WAGR has also been an EU-supported pilot-dismantling project and Nuclear Electric and Scottish Nuclear made significant financial contributions following the 1993 review.

Decommissioning plan

The principal technical exercise is the dismantling of the reactor core and pressure vessel containing neutron activated components with dose rates well in excess of 1Sv/hr. The reactor had a neutron shield to reduce the dose rate on the operating floor and this permitted some semi-remote and manual dismantling operations to take place above it. However below the neutron shield all work had to be carried out remotely.

The components installed for remote dismantling of the reactor core comprised:

- A remotely operated dismantling machine to deploy tools.
- A recovery and transport system.
- A shielded waste route through which to move, sort, assay and package the waste.
- A conditioning plant where waste is treated.
- A storage/disposal container.
- An interim storage facility for Intermediate Level Waste.

Preparatory work

The first years of the decommissioning project were spent designing the necessary components, modifying the structure and installing the equipment. A schedule of the preparatory operations is shown below:

1981	The Windscale AGR was shut down.
1983	Initial decommissioning began and fuel removal was completed. Associated materials were removed and treated ready for disposal.
1984-1988	The waste route was constructed.
1989	The reactor refuelling machine was dismantled.
1990-1992	The reactor's top biological shield and pressure vessel top dome were removed and a new heavy duty removable shield installed.
1993-1994	The remote dismantling machine was installed and the interim intermediate level waste store constructed.
1994-1995	The four 190 tonne Heat Exchangers were lifted through holes cut in the containment building using a 1 200 tonne capacity mobile crane and transported by heavy duty low loader for disposal at the low level waste repository.

1995	The waste route was finalised and the encapsulation plant installed.
1997-1999	Non-active commissioning was performed demonstrating the safe and efficient operation of the installed equipment.

The principle methodology for remotely dismantling components from the reactor core may be explained as follows:

- Waste is released by campaign specific tools deployed from either the 3 Te transfer hoist or the manipulator mounted on the manipulator platform.
- Waste items either individually or grouped in waste box furniture are moved from the reactor vault into the adjacent sentencing cell.
- Waste items either individually or grouped in waste box furniture are lowered from the sentencing cell into the upper loading cell for waste assay. Gamma dose rate measurements are applied to Waste ESTimation (WEST) codes and the inventory calculated.
- Waste is lowered from the Upper Loading Cell into the WAGR box in the lower loading cell.
- Full boxes are driven into the concreting cell where the waste is encapsulated in cement grout and the box completed with an in-situ cast reinforced concrete lid.
- Boxes are moved to the transfer cell, unwrapped, monitored and moved to the weigh station to cure.
- Boxes are despatched for interim storage (ILW) or disposal (LLW)

Reactor core and pressure vessel dismantling – the campaigns

The reactor core and pressure vessel is being dismantled in a series of 10 campaigns each linked to a reactor component. A final campaign to complete the clean-up of the vault space has recently been conceived as an eleventh campaign. These are shown in figure 4 and briefly described below:

1 Preliminary Operations:
   Controlled manual operations to prepare the top of the hot box and surroundings for remote operations.

2 Operational Waste:
   Removal of ILW operational waste temporarily stored in the fuel channels. E.g. Arrestor mechanisms, control rods, neutron shield plugs, blockers, specimen containers and sample pots.

3 Hotbox:
   The 31 Te gas manifold with a multitude of internal components used to divert the hot coolant gas into the 4 heat exchangers.

4 Loop Tubes:
   The 6 highly activated (~120Sv/hr) stainless steel tubes located in the reactor core to form experimental fuel channels.

5 Neutron Shield:
   A flat cylinder constructed of 3 layers of graphite blocks with stainless steel fuel channels. In all, 2 300 components weighing 80 tonnes.

6 Graphite core and restraint structures:
   200 tonnes of graphite blocks in 8 layers, with fuel channel and reflector blocks restrained by a tensioned steel beam and interlaced with instrumentation.

7 Thermal shield:
   A 6” (150mm) thick cylinder surrounding the core constructed of interlocking steel bricks held together by fishplates.

8 Lower structures:
A steel ring beam attached to the pressure vessel supporting a diagrid and plates to support the weight of the core components and transfer the load to the pressure vessel

9 Pressure vessel and insulation:
The reactor pressure vessel below the neutron shield, insulated with asbestos blocks clad in aluminium sheet.

10 Outer ventilation membrane and thermal columns
A steel liner to the concrete bioshield and two graphite thermal columns

11 Clean up and asbestos clearance:
A new campaign to clean the reactor vault and waste route to enable the area to be cleared of the asbestos licence requirement.

Safety management

The decommissioning operations are covered in a Decommissioning Safety Case (DSC) based on the decommissioning plan and covering all the bounding case hazards. Each campaign gains safety approval by submission of a specific campaign safety case categorised in accordance with the UKAEA procedure and following the appropriate route for gaining Authority to Proceed.

Safe systems of work are provided in the form of Campaign Driving Instructions (CDIs). These are detailed method statements for each campaign and are backed by operating instructions and risk assessments. CDIs are written by the operators and approved by the Duly Appointed Person (DAP) on behalf of the licensee.

The dismantling campaigns - progress

The first 9 campaigns have been completed and Campaign 10 is currently underway:

Campaign 1 - preliminary operations

This was a programme of manned access to the top of the hot box and surroundings to prepare for remote operations. Work included removal of debris from previous dismantling activities, repairs to damaged fuel channels to permit access for grabs to recover operational waste, replacement of hot box permanent supports with temporary ones and cutting and removal of the co-axial gas coolant ducts.

Campaign 2 – removal of operational waste

The operational waste was removed using the 3 tonne hoist and lifting grabs designed to engage with the pintels fitted to each waste item. Where pintels were damaged disposable rare earth magnets were used to achieve the lift. Each box contained furniture to hold 110 items of operational waste and in all 770 items were removed from the reactor. Some of the waste items had significant activity and hence high-density boxes were used.

Campaign 3 – hotbox

The hot box was dismantled in a series of mini campaigns using 40 - 200 amp Plasma torches deployed both by remote rigs and used manually. Despite extensive use of full size mock-ups, a number of difficulties with the deployment systems were encountered however such difficulties were overcome by carefully controlled manual intervention. In one case remote equipment failed to carry out the task as the hotbox installers had added temporary fixings and failed to either remove them or mark them on the as-built drawings. In total, 9 different plasma torch deployment systems, 1 remote shear, 4 different grabs and a Slingsby manipulator were used together with cutters and hand tools when manual intervention was employed.

Campaign 4 – loop tubes

The tubes were filled with high density grout by pumping from below, and then cut into sections using a remotely deployed 750 tonne shear before being assayed and packed in waste box furniture designed to provide maximum shielding by the encapsulating grout.

Campaign 5 – neutron shield

The neutron shield was removed in a series of 11 mini campaigns – 103 tonnes of graphite and steel were removed and packaged in 32 WAGR boxes, (22 LLW and 10 ILW). Problems were encountered in cutting the stainless steel liner tubes due to hardening and embrittlement in reactor conditions.

Campaign 6 – graphite core and restraint structures

The graphite core consists of 200 tonnes of graphite blocks in 8 layers and many tools employed in the neutron shield campaign have been re-used.

Each layer is restrained by a tensioned steel beam which was cut into sections using a standard industrial quality reciprocating saw mounted in a remotely deployed frame, fitted with additional motor systems to advance the blade and clamp the tool in position. Remotely deployed clamps were designed to hold the components in place until the saw cut was completed.

The fuel channel blocks are stacked vertically and were removed using a ball grab. The reflector blocks were lifted using a drill tool suspended from the hoist that first drilled three holes in the block, then tapped a thread to which the tool was attached. After lifting and packing, the motor direction was reversed and the block released from the tool.

The final activity to complete core removal was to cut the gas baffle attached to the bottom of the thermal shield. After using the manipulator with a yard brush attachment to clean the surface of the gas baffle, the manipulator was again used to hold a 40 amp plasma torch to the cutting position and the RDM rotated to perform the cut.

The whole campaign was a complete success with no significant problems encountered. The waste was encapsulated in 59 WAGR boxes.

Campaign 7 – thermal shield

A 6” (150mm) thick steel cylinder surrounding the core constructed of 168 bricks stacked in 14 courses and held together by fishplates. Each 1.3 tonnes brick is constructed from three 2” thick plates bolted together such that the edges lock together. The internal face of the thermal shield was crossed with thermocouple wires and flux scanning tubes effectively joining adjacent bricks. These were severed at the brick interfaces using a manipulator mounted shear. The design concept was then to remove the fishplates that hold the bricks together, and using the original lifting features remove the plates one by one using the 3 Te hoist. There was a significant risk that corrosion or distortion would cause the plates to bind together and a series of tools were designed to cover all eventualities:

- Simple lifting beam to attach to existing features.
- Complex lifting beam using 3 Camlock devices to attach to the top of the brick (in the event that the existing features were inaccessible or unsafe).
- A jacking frame to work with either of the above grabs (to exert up to 10 Te in the event that the pull from the 3 Te hoist was insufficient to remove the brick).
- A nut-runner tool to unbolt the inner of the three plates and release grip.
- A gas torch to remove the edges of the inner plate to release grip.

The last two items were not taken beyond concept design as information from drawings and from an individual who had been involved in the construction of WAGR gave greater confidence in the basic design.

Removal of the thermal shield proved to be straight forward and the jacking frame was not required to be deployed.

Ventilation plant upgrade

In order to cope with the requirements of hot gas cutting within the reactor (removal of lower structures and pressure vessel) an upgrade of the reactor ventilation plant was undertaken. Pre filtration was to be provided by momentum separator, spark arrestor and cyclone before the airflow was passed through HEPA filtration to discharge. Discharge monitoring systems for particulate activity, gaseous tritium and carbon 14 (whilst cutting the pressure vessel) were transferred from the previous system. The work was programmed for completion to coincide with the oxy propane cutting in Campaign 8.

Campaign 8 – lower structures

For safety clearance reasons this campaign was divided into two sections: a) those elements with a similar risk envelope to campaign 7 and b) those that require the introduction of gas cutting.

a) These are components that required the development of specialised cutters and pullers to remove them: arrestor mechanism housings, reactor core bearing lower races, neutron baffle plates, the bottom layer of thermal shield plates and the core support plates. All were removed without difficulty.
b) The Diagrid and Ring Beam: The diagrid (914 mm deep and 57 mm thick) was removed by oxy-propane cutting in a sequence to retain the stability of the ring beam. The torch was mounted on a vertical motorised track that hung from the diagrid and was deployed by the 3 Te hoist. The torch was fitted with safety cut-outs to ensure that the gas supply was cut off if the flame failed. Detectors were fitted to the ventilation system to warn against build up of either propane or oxygen. After deploying the torch tracking system, the 3 Te hoist deployed the appropriate grab and the cut was made. Each piece released was removed and placed in waste box furniture.
c) The ring beam, a hollow rectangular section 254mm wide and 610mm deep, was to be cut using an Oxy-Propane with Iron Powder Injection (OPIPI) system to sever the section in one pass. The ventilation filtration system proved inadequate and blinded very quickly. To maintain progress a revised system that used oxy-propane cutting in two passes to complete each cut was adopted to minimise filter blinding. Following ring beam removal the ring beam support bearings were removed using a reciprocating grab.

The 70 Te of waste from Campaigns 7b and 8 were packaged into 12 ILW boxes and 3 LLW boxes.

Campaign 9 – reactor pressure vessel (RPV) and insulation

The steel RPV was 6.5m diameter and 73mm thick with a hemispherical bottom all insulated with 216mm of asbestos insulation, a layer of asbestos rich render and clad in aluminium. The RPV was supported from the corbel by 20 angled steel strakes that transferred the load into the bioshield structure. New supports had been installed in the 1980s within the crypt below the RPV but the load had not been fully transferred to the new supports.

The removal of the RPV was planned as a series of mini campaigns:

1	Install shielding/physical barriers	Remote
2	Trim top edge of RPV	Manual
3	Cut and remove RPV support strakes	Manual
4	Transfer RPV load to new supports by progressive heating of last 3 strakes	Manual
5	Remove insulation boxes between RPV and concrete corbel	Manual

Manual access was acceptable as the dose rate had reduced to 60 - 90 μSv/hr

6	Upper barrel section - uninsulated	Remote
7	Lower barrel section (part)	Remote
8	Tundish	Remote
9	Lower barrel section (part)	Remote
10	Gas baffle	Remote

11	Central catchpot	Remote
12	Lower Hemisphere	Remote

Upper barrel section

The upper barrel section was cut using an oxy-propane torch. Two torch deployment tracks were used, one for vertical cuts and one for horizontal cuts. The waste sections were moved into the sentencing cell, placed in furniture, assayed and encapsulated in WAGR waste boxes.

Lower barrel section

The methodology assumed that the RPV and insulation would act as a composite and could be removed in sections. The cutting technology adopted was OPIPI in order to cut the composite section in one pass from inside the vessel. Because the insulation was asbestos based, all work had to be in compliance with the Control of Asbestos at Work Regulations. Because of the radioactive nature of the work, it was also subject to the Ionising Radiations Regulations 1999. Advice from the Health and Safety Executive (HSE) was that all operators disturbing asbestos should be licensed asbestos workers. Thus all the operators who had been specifically trained to operate the WAGR plant and equipment, were re-employed through a contractor who holds an asbestos licence.

Work proceeded on the removal of the lower barrel section using OPIPI cutting deployed on the same two track systems. Two problems were immediately apparent:

- Excessive blinding of the HEPA filters in the reactor ventilation system
- The insulation was not attached to the RPV and was not a composite waste form.

In investigating the filter issue both thermal cutting techniques and the ventilation system were reviewed. It did not prove possible to improve the cutting techniques and improvements to the ventilation could not be identified at justifiable cost. Thus a case was made to continue with this methodology. The high cost of disposal of the large number of asbestos contaminated filters was improved by the development of a local compaction process using medium force compaction giving a volume reduction factor of 3.

Remotely deployed clamps were produced to hold the composite waste forms in one piece during cutting and box packing.

Tundish

To maintain progress, the sequence of activities was reviewed and the removal of the tundish accelerated to be undertaken whilst the cutting reviews were progressed. Clearance of debris and insulation blocks from previous operations was completed in September 2006 and cutting began in October 2006 using oxy-propane torches mounted in deployment machines designed to produce radial and circular cuts. UKAEA also began development of a manipulator mounted torch with significant benefits over the tracked/machine deployed ones and this was tested towards the end of the tundish removal.

Gas Baffle

Following completion of the RPV barrel section, the Gas Baffle, a lining to the RPV lower hemisphere was successfully removed using the manipulator mounted torch.

Central Catchpot

The upper section was removed using the manipulator mounted torch. This exposed the contents, a mixture of operational and decommissioning debris which were removed using various modified grabs suspended from the 3 Te hoist and a venturi vacuum cleaner system.

Lower Hemisphere

To avoid the problems associated with OPIPI cutting a revised methodology was developed using the manipulator mounted oxy-propane torch. Firstly the steel was cut into small pieces and removed using a bucket grab. Bulk insulation was then removed and the aluminium cladding cut into sections with the torch. Much of the waste produced fell to the floor of the reactor vault and was later collected and packaged.

Campaign 9 was completed in March 2010

Campaign 10 – outer ventilation membrane and thermal columns

This commenced in May 2010 following a brief shut down to reconfigure areas of the plant. At the time of updating this report the work was progressing satisfactorily.

A1.2 Category 2 reactor projects (dormant)

A1.2.1 Gentilly-1, Canada

Gentilly-1 was a heavy water moderated, direct cycle, boiling light water cooled prototype reactor that was shut down in 1979 after 7 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 and in the reactor building. Asbestos insulation removed from the turbine building during the initial decommissioning activities was shipped off site. The roofing of the turbine building has been replaced.

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 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$ 600 000 (2010).

Unique attributes/challenges

The G-1 facility remains in a storage with surveillance state and no decommissioning activities have taken place. No unique attributes or challenges have been identified to date that would affect the final decommissioning activities.

New methodology developments

None

New equipment or instrumentation developments

None

Licensing and political issues

No significant issues identified to date

Lessons learned

None

A1.2.2 Nuclear power demonstrator (NPD), Canada

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

During facility shutdown, all nuclear systems were drained and sealed. All fuel was shipped off site for dry storage in concrete canisters at Chalk River Laboratories. Operational wastes were shipped to Chalk River for disposal. At the beginning of the "storage with surveillance" state, there were approximately 2×1 015 Bq on site, mostly in the reactor vessel.

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

The NPD decommissioning project to implement the "storage with surveillance" state was budgeted at C$ 18.5 million. Since 1988, the facility has been in the "storage with surveillance" state during which the facilities, remaining operating systems and any remaining radioactive areas have been routinely monitored and maintained. Site security is monitored from Chalk River Laboratories and complemented with periodic on-site inspections. The annual storage with surveillance costs are of the order of C$ 0.4 million compared to C$ 14 million before decommissioning.

Unique attributes/challenges

The NPD facility remains in a storage with surveillance state and no decommissioning activities have taken place. No unique attributes or challenges have been identified to date that would affect the final decommissioning activities.

New methodology developments

None

New equipment or instrumentation developments

None

Licensing and political issues

No significant issues identified to date

Lessons learned

None

A1.2.3 Rapsodie, France

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

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

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

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

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

Project has been stopped from 1994 to 2005.

The safety files to restart has been send to the ASN in 2008, decommissioning Decree is expected in 2011.

A1.2.4 G2/G3 Reactors, France

G2 and G3 were two 250 MWth gas-graphite reactors that operated between 1958 and 1980. In each reactor, the core with reflector and shield plates is located within a pre stressed 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 37 M€. After studies, it was decided to wash the interior of the systems with high-pressure water and then melt the piping for further recycling the metal expecting an exemption level in France.

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

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

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

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

The reactors G2 and G3 have formally been placed in a Stage 2 dormancy status. The dormancy period is expected to be between two and three decades.

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

A1.2.5 Vandellos 1, Spain

The Vandellos 1 plant was a 460 MWe gas graphite reactor of the same type as the EDF St-Laurent-des-Eaux plants. It was shut down after 17 years of operation, after a fire in the turbines in 1989, a level III incident. There was no release of radioactivity, but the fire destroyed the conventional plant and flooded the bottom part of the reactor building.

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

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

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

The graphite fuel sleeves, which were stored in 3 silos on site, consisted of about 1 000 t of graphite and 2 t of stainless steel wire, with the graphite containing low activity, long life 14C and the wire containing high activity, relatively short life 60Co. 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. By June 2003, the goals of the project had been reached, starting the dormancy period.

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 60Co. 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/cm2 and to evaluate the leakage over a period of time. The results were very satisfactory, about 18% of the acceptance criteria. This test has been repeated in 2005 with similar satisfactory results.

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

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

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

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

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

The end of the level 2 project has been reached in the second quarter of the year 2003. The total budget of this level (fuel and waste disposal not included) was about 96 M€.

A1.3 Category 3 reactor projects (complete)

A1.3.1 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 8.6×1 012 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/cm2 for β- and γ-emitters;
- 0.37 Bq/g for β- and γ-emitters;
- α-emitters were not encountered during the intensive characterisation work that has been done determining the key nuclides of each decommissioning phase.

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

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

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

A1.3.2 Heissdampfreaktor HDR, Germany

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

The aim of the project was to completely dismantle the facility to establish "green field conditions" and was executed under 3 subsequent sub-licences. Due to the short operating life of the reactor, the activity inventory was small (about 2.2×1010 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/cm2 for 137Cs.

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.

A1.3.3 Japan power demonstration reactor (JPDR), Japan

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

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

The decommissioning project was conducted in two phases:

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

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

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

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

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

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

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

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

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

- The ratio of manpower expenditure to the weight of dismantled components was evaluated to be 500-2 000 man-hours/t in remote dismantling procedure for highly radioactive components, compared to 10-100 man-hours/t with manual dismantling procedure in the reactor building. The remote dismantling systems were proved to be effective for general components to minimise the radiation exposure of workers, which was kept to a collective dose of approximately 300 man-mSv.
- After the removal of the components and systems, the inner surfaces of the JPDR buildings were decontaminated using a number of techniques, including scabblers, needle guns and concrete planers. The total area to be decontaminated and surveyed (radiologically) before release is 12 000 m2. 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/cm2 for 60Co contamination in 60 seconds counting time. Two kinds of detectors were fabricated and these were attached to the movable machines for measurement of radioactivity on building surfaces and piping embedded in building structures. As for remote dismantling techniques, dual arm manipulators were manufactured to study on automated remote dismantling work based on computer simulations. The dual arm manipulators are controlled by the packages of robotic language prepared by computer simulations. The applicability of automated control system was examined by dismantling mock-ups of components. In systems engineering, project management tools using expert systems and database on dismantling activities have been developed. The systems are intended to be applied to estimation of radioactive inventory, project resources, worker dose, and scheduling in a decommissioning project by referring data obtained in past experience.

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

A1.3.4 Shippingport, United States

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

The objectives of the decommissioning project were to:

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

Conceptual and detailed engineering for the decommissioning project was completed in 1983. The plan was to decommission the plant to Stage 3. The physical decommissioning consisted of the demolition and disposal of 26 various fluid and electrical systems before the buildings could be demolished. In all, about 17 100 m of contaminated piping and 16,800 m of non-contaminated piping, and 1300 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 6.14×1 014 Bq, of which 6.09×1 014 were contained in the reactor vessel. Total project radioactive waste volume disposed of was 6 057 m^3 weighing approximately 4185 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 the 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 8400-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.

Lessons learned

The main lessons learned from the project were:

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

A1.3.5 Experimental boiling-water reactor (EBWR), United States

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.

A1.3.6 Fort St. Vrain, United States

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

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

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

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

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

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

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

- The graphite from the reactor vessel – 1 770 pieces with surface dose rates up to 3 Sv/h – was sent off site as low-level waste.
- The upper core barrel (9.15-m diameter, 8.85-m height, 67-mm wall thickness) was segmented under water with a remotely operated plasma arc torch. The segments were shipped to Hanford as low-level waste.
- Two shifts of eight divers, each diver making a 90-minute dive, were utilised:
  - First to clean up debris, etc., from the core support floor.
  - To use underwater jack hammers to remove silica plugs in the core support posts.
  - To use a remote plasma arc cutting tool to free inconel sleeves in the posts.
  - To use handheld plasma arc torches to remove the stainless steel floor thermal seal.
- The inconel sleeves had contact doses of 200-500 mSv/h and so steel work platforms were designed to keep the divers at a safe distance.
- The core support floor was cut loose from its supports during 1 250 dives, performed over ten months. A collective dose of 173 man mSv was taken.

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

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

A2.1 Fuel facilities and fuel related projects in progress

A2.1.1 Eurochemic plant, Belgium

Introduction

The Eurochemic reprocessing facility at Dessel in Belgium, was constructed from 1960 to 1966. A consortium of 13 OECD countries operated this demonstration plant from 1966 to 1974, and reprocessed 180 tonnes of natural and low-enriched and 30 tonnes of high-enriched uranium fuels. After shutdown, the plant was decontaminated from 1975 to 1979 to keep it in safe standby conditions at reasonable cost.

In 1984, Belgoprocess took over the activities on site. When it was decided in 1986 not to resume reprocessing in Belgium, the main Belgoprocess activities changed to processing and storage of radioactive waste and to decontamination and decommissioning of obsolete nuclear facilities.

The industrial decommissioning of the main process building of the former Eurochemic reprocessing plant was started in 1990, after completion of a pilot project. Two small storage buildings for end products from reprocessing were dismantled to demonstrate and develop dismantling techniques and to train personnel. Both buildings were emptied and decontaminated to background levels. They were demolished and the remaining concrete debris was disposed of as industrial waste and green field conditions were restored.

The main process building is a large rectangular construction of about 80 m long, 27 m wide and 30 m high. The core of the building consists of a large cell block of 40 main cells, containing the chemical process equipment. Access areas and service corridors are located on 7 floor levels. About 106 individual cell structures have to be dismantled. Some cells have contamination levels up to 125 Bq/cm² (beta) and 200 Bq/cm² (alpha). Some hot spots give a gamma dose rate of several mSv/h.

About 1 500 Mg of metal structures, and 12 500 m³ of concrete with 55 000 m² of concrete surfaces have to be removed and/or to be decontaminated.

The specific Belgoprocess approach should be highlighted, considering that

- The decommissioning activities are carried out on an industrial scale with special emphasis on waste minimization, extensive decontamination to unconditional release levels and cost minimization;
- Commercially available technology is used in good co-operation with the nuclear or non-nuclear industry;
- The decommissioning of a nuclear power plant mainly being characterized by radiation risks due to activation- and fission products, the alpha contamination on equipment and building surfaces in a reprocessing plant requires the use of adequate protective clothing.

Overview of decommissioning activities and equipment used

- Dismantling of metal components is mostly carried out by plasma-arc cutting. Pipes were also cut with radio-controlled hydraulic shears, while dry or wet cutting of cast iron shielding blocks is done with hydraulically controlled saw blades.
- Cutting and decontamination of concrete structures is carried out either hands-on, or by electrically powered, hydraulically controlled systems. Mini electro-hydraulic hammering units are used when contamination has penetrated deeply into the concrete

surface, increasing the decontamination possibilities and reducing the workload for the operators. Cell entrances are created or enlarged with diamond cable cutting machines.

- In the early days, concrete walls with limited in-depth contamination, were decontaminated using commercially available pneumatic hand scabblers. To improve the working conditions for the operators, and to increase capacity, scabblers were progressively automated. Operation efficiency was improved when shaving machines were introduced, using a diamond tipped rotary head, designed to give a smooth surface finish and making monitoring easier. Due to the absence of machine vibration, the physical load to the operators was also reduced.
- Pipe penetrations between the cells have been removed after the decontamination of the walls and ceilings and before the first release measurement. These internally contaminated pipe penetrations were closed and welded before removal. For the removal of the pipe penetrations a concrete splitter together with pneumatic hammering was successfully applied.
- To remove process equipment in a safe and ergonomic way from cells with heights up to 18 meters, movable platforms are used. On the movable platform a video survey system is installed, enabling the two operators in a cell to be monitored by an operator outside the work area. A communication system provides radio contact with the operators on the working platform. In other cells, lifting platforms with articulated axes are used.
- To decontaminate contaminated profiles and plates on an industrial scale a dry abrasive blasting machine was developed. Operational activities started in 1996 and at the end of December 2009 about 1 314 Mg of contaminated material has been treated. About 1 011 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 310 Mg of concrete and heavy concrete blocks were decontaminated. 273 Mg 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.
- The operators use various combinations of protective clothing and equipment, especially in areas with alpha contamination. To provide breathing and cooling air to the operators in their protective clothing, a specific personal protection system was developed.
- As in the metallurgical industry, in construction, and in forestry, exposure to hand-arm vibrations also occurs in the decommissioning of the nuclear installations. Therefore Belgoprocess has set up a global representative valuation method. The results of the analyses carried out do not give reasons to some concern.

Release of decontaminated materials

Release of decontaminated material is based on current procedures, which means that all equipment, material and areas with contamination levels above background are considered radioactive. Surface area has to be monitored 100%, and surfaces/areas that cannot be monitored are considered radioactive.

A specific approach was developed for taking representative samples and monitoring concrete material in view of the final demolition and unconditional release of the remaining structures of the various buildings after dismantling and decontamination.

For the small buildings in the pilot project, all concrete surfaces were monitored twice in view of unconditional release, and core samples were taken at the previously most contaminated places. For the remaining structures or larger buildings, this will result in a large number of samples to be taken and to be analyzed. In addition, it will be very difficult or impossible to prove that these samples are representative for the remaining structures of the buildings.

For the main reprocessing building an alternative release procedure has been developed, considering at least one complete measurement of all concrete surfaces and the removal of detected residual radioactivity. This monitoring sequence is followed by a controlled demolition of the concrete structures and crushing of the resulting concrete parts of the cells to smaller particles. During the crushing operations, metal parts are separated from concrete and representative concrete samples are taken, the frequency of sampling meeting the prevailing standards. In a further step, the concrete samples are milled, homogenized and a smaller fraction is sent to the laboratory for analyses. After approval of the licensing documents, operations of the facility were started in June, 2001.

At the end of December 2009, 4 018 Mg of concrete were monitored. All this material was unconditionally released and removed from site after analyses and agreement by the in-house health physics department and the authorities. The material is further used in conventional road construction.

Demolition of the decontaminated main process building

To facilitate demolition, the main process building was beginning 2004 divided into three parts. Each part is isolated from the others and each section (eastern, central and western parts) will be demolished individually. The demolition of the fully decontaminated eastern part started in June 2008 and was completed in September 2008. During the demolition of the eastern part decommissioning activities in the remaining and separated building were continued. On May 17th, 2010 the demolition of the central part will start. So further decontamination works can be finalized in the remaining western part waiting for final demolition in 2012-2013.

Future decommissioning programme

The decommissioning operations carried out at the main building of the former EUROCHEMIC reprocessing plant have made substantial progress and will be finalised into a few years.

Additional installations will be put in a standby status on the EUROCHEMIC site during the next years, and will become available for decommissioning. It considers buildings 105, 122, 121, 124, and others, presenting the specific challenge to dismantle the large storage vessels for the high level liquid wastes from the EUROCHEMIC reprocessing activities, and still containing an important amount of radioactive material.

Also on the site of the former waste treatment department of the Belgian nuclear research centre, an important number of installations will be put in a standby status, available for decommissioning. It considers a number of storage facilities, the incinerator for beta-gamma wastes and some water treatment installations

A2.1.2 Building 204 bays decommissioning project, Canada

The 204A and 204B Bays are storage pools located in Building 204 (B204) at Chalk River Laboratories, Canada, and are associated with the NRX Reactor operation. They were installed 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 from NRX to the Building 220 (B220) fuel processing facility. After alterations in 1958-59, they were separated by sand-filled sections of trench and concrete dams into two areas defined as 204A and B bays. The 204A Bays remained operational until the NRX Reactor shutdown in 1993. The 204B Bays have remained water filled but unused since 1959.

When the project joined the Co-operative Programme, the 204A Bays contained water, approximately 8 m3 of sludge and algae, and operational components. The sludge, algae and water have since been removed through a vacuuming and pumping process. All equipment and tooling has been removed from the 204A bays. The 204B Bays contain water and a build-up of sludge and algae but no operational components.

204A Bays

Water	800 m^3 capacity		Water removed
Components, etc.			Components removed

204B Bays			
Water	400 m3	3.4×1011 Bq	75% 137Cs, 20% 90Sr, 5% other
Algae/sludge, etc.	9 m3	9.0×1010	88% fission products, 10% actinides
Components, etc.	None		

The objective of the B204 Bays project is to clean out the Bays, at present containing algae/sludge, and equipment and store them in a dry state, with only fixed contamination and requiring minimum maintenance, monitoring and surveillance. At present, the 204A Bay is drained and a historical leak of 4 m^3 of water per day has stopped.

Detailed decommissioning plans were submitted to the regulators for both 204A 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 required significant effort 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 Environmental Assessment was approved by the regulator in 2006 October. In preparation for physical work, two Advanced Decommissioning Work Packages (ADWP) were submitted to the regulator and approved to address a directive from the regulator to deal with a fire safety issue in the bays. The first ADWP was approved in 2006 March to remove the water from the 204A bays and the second one in 2006 September to remove approximately 35 meters of wooden structure between NRX reactor and the B204A bays superstructure to create a firebreak. Both work packages were completed by the fall of 2008 and the connecting concrete fuel bay trench between NRX reactor and B204 bays was sealed and covered with weather proof covers and left in place for future removal and under Phase 2.

Work is continuing to prepare for cleaning the 204B 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) will have to be designed and manufactured for this portion of work. The Detailed Decommissioning Plan is being revised to be in line with the current strategy of decommissioning the bays and attached fuel processing facilities.

Unique attributes/challenges

B204 network of bays consists of unlined painted concrete trenches. In the 1950s the bays were divided into B204A and B204B bays by adding concrete dividing walls, emptying large sections of the bays and backfilling these sections with sand. The concrete dividing walls have leaked through over the years and the water level in the 204B bays and the sand filled trenches is at the same level. This has resulted in a large volume of contaminated soil to be removed from the concrete bays. The bay circulation system was also cut off by separating the 204 A&B bays resulting in stagnant water that has been left uncovered for approximately forty years. Dust and debris has gathered in the bottom of the bays.

A bay circulation system was also added to the 204A Bays in the same time frame by adding a PVC drainage system on top of the existing floor and covering with a new concrete floor. The PVC pipe has deteriorated and has become brittle resulting in fatigue of the pipe. Decommissioning will involve cleaning of the drainage pipe that has many low areas that may have collected failed fuel particles and removal of two layers of concrete with contamination embedded between them.

New methodology developments

No new methods or developments were used in the emptying of the bays and removal of the structure over one section of the 204A bays.

New equipment or instrumentation developments

No new equipment or instrumentation was developed to date for this work. Off the shelf equipment has been used to date.

Licensing and political issues

The regulator requires a revised Detailed Decommissioning Plan to be submitted to reflect the current strategy for continuing with the decommissioning of the 204 bays. No further work can progress until this document is approved by the regulator.

Lesson learned

Numerous lessons were learned during the completion of the two Advanced Decommissioning Work Packages. Some of these lessons are listed below:

- Under water radiological surveys need to be accurate and confirmed before water is removed. A faulty piece of equipment was used for all the surveys even though it was calibrated and verified for use along with a second instrument that was giving the same readings. Although the readings taken were in the proper range, it was determined during water removal and the higher than expected fields during the last meter of water removal that the underwater surveys were inaccurate.
- Cleaning of algae and sludge from the bays is very time consuming and can lead to issues trying to meet ALARA. Bringing highly contaminated sludge to the surface to dry before being placed in waste cans was a risk to workers.
- Use of personal protective clothing and equipment needs to be well thought out at the beginning of work to prevent dose uptakes by the workers. Constant vigilance in this area resulted in no contamination events.
- Fatigue and dehydration needs to be considered during this type of work while dressed in their protective clothing.
- Dress/undress areas need to be well maintained to prevent the spread of contamination and all workers should be properly trained on protocols associated with them.
- Removal of one hazard can uncover another. During the removal of water from the bays, a modification to a service pipe was required to relocate it in preparation for the structure to be removed. Delays in completing this mod coupled with the advancement of water removal resulted in a six month delay to project due to high fields as a result of water removal in the area where the service pipe had to be tied in. Extensive shielding had to be installed to protect the workers performing the tie in.
- Following the removal of water from the bays and the fire separation completed, the bay walls and floor have been allowed to dry which has resulted in peeling paint and increased contamination in the bays. Sealing the concrete walls should be done as soon as possible following the water removal.

A2.1.3 Retrieval and Management of Legacy Waste at the Ispra Joint Research Centre of the European Commission

Background

The European Atomic Energy Community, empowered in 1957 by the Euratom Treaty, contributed to the growth of the pacific use of nuclear energy in the European Union with the creation of a series of Joint Research Centres (JRC). The nuclear activities started in the late 50's and were conducted for about twenty years. Afterwards, they were progressively reduced, and the original nuclear installations have been shutdown and are currently being held in "safe conservation". The JRC is required to decommission the nuclear installations and manage the associate waste.

To achieve this scope, in agreement with the European Parliament and the Council, a "Decommissioning and Waste Management Programme" (D&WM) has been put into a action. The D&WM Programme embraces the majority of JRC nuclear installations, most of which are in the Ispra site. The decommissioning of the JRC Ispra site nuclear liabilities will be accomplished in the next two decades (2010-2028), with the ultimate objective of a possible conventional re-use of land and buildings (stage 3, as recommended by IAEA).

The research activities have led to the accumulation of a certain amount of operational waste. In some cases the waste has been conditioned and disposed underground and, for this reason, it must be retrieved and appropriately reconditioned.

The waste management infrastructures include the characterisation, treatment and storage facilities present in the so-called "Area 40" and some appendices such as a Tank Farm Facility (TFF) and a POCO buffer.

The projects accepted and included in the OECD CPD deal with the retrieval and management of the waste disposed in the "Bituminised Drums" and in the "Roman Pits", and of the small quantities of ILLW and sludge present on site.

Bituminised drums management

The Bituminised Drums contain various combinations of VLLW and LLW and inert matrices originated from routine nuclear activities conducted on site. Three 50 m long trenches were completed between 1966-1988, and host about 6 500 standard 220 litres drums, for a total raw waste volume of around 1 230 m^3. The majority of the estimated 500 GBq total activity is in the newest trench, and is mostly due to Cs-137, although the presence of Co-60, Ni-63 together with other radio-nuclides (some of which α) cannot be excluded. Surveillance wells in proximity to the pits have indicated no evidence of contamination of the underlying water table.

The drums must be recovered from their present location, characterised and reconditioned. Given the novelty of the task, a sample validation test will be performed to demonstrate their safe recovery. Various options have been considered for the drums treatment ranging from containerising and immobilising the bituminised drums in larger drums or containers, to incinerate them to produce a vitreous-metallic waste or to separate waste from bitumen and manage them separately.

JRC Ispra has chosen as reference strategy to transfer and immobilise the drums without treatment in 5.2 m^3 prismatic containers (CP-5.2 according to Italian UNI Norms). The estimated volume of conditioned waste, repackaging 8 drums per container, would be around 4 200 m^3 (half of the expected conditioned LLW/ILW volumes produced).

The main steps of the reference strategy are the followings: extraction, wrapping and brushing, sorting of homogeneous batches with an X-Ray Digital Radiography System (XDRS), performance of the necessary Destructive Analysis (DA), characterisation, containerisation in CP-5.2, grout with cement, final characterisation, label and transfer to the Interim Store Facility (ISF). The management cost has been estimated in 13 M€, which includes the containers provision (5 M€) and the XDRS facility provision (1,25 M€) while it excludes the final repository costs (which depend on the feasibility to reduce the drums initial volume).

The management strategy implies the provision of an environmental cover and various drum handling equipment in order to extract them in an environmentally safe enclosure. It has been estimated that about 2 000 drums per year can be characterised.

If during recovery contaminated soil is encountered, according to a feasibility study, a nitric acid leaching based technique would be the most effective for the removal of Sr-90 and Cs-137 whereas leaching with a Ce saturated solution would be effective for Pu-241 removal. These processing steps could permit the conditional release of about 90% of the soil. The proposed treatment steps would require an investment of about 890 k€ for the design and construction of a dedicated facility and about 2 k€/tonne to operate it. Based on the estimation of 150 t of contaminated soil, the total decontamination costs would be of 1.19 M€, i.e. 13 k€/m3 which exceeds the cost of soil super-compaction or even the cost of directly packaging the soil.

The main project milestones are the procurement of the environmental cover (by 2Q 2012), the drums extraction validation test (by 1Q 2013), the Safety Authority approval of the extraction operational plan (by 2Q 2013), the drums extraction, treatment and conditioning (between 2013-2017) and the site remediation (by end of 2017).

1) Unique Attributes/Challenges: Retrieval, characterisation and re-conditioning of different radioactive waste streams in non standard bitumen matrices; historical data validation.
2) New Methodology Developments: Treatment and reconditioning.
3) New Equipment or Instrumentation Developments: None, nuclear use of conventional equipment like shredders, plasma torch etc.
4) Licensing and Political Issues: Qualification, licensing and acceptance by the operator of the future Italian National repository of Final Waste Packages containing organic waste.
5) Lessons Learned: Bitumen matrix quite stable: no contamination of the underlying water table detected after more than 30 years even in poor conditions; in consequence no large soil or water contamination expected during the extraction;

Roman pits management

The fifteen Roman Pits are underground pits built between 1965-1978 for the shallow land burial of solid LLW/ILW. They are constructed from prefabricated concrete rings, ~1.35m diameter, superimposed one another to reach a total height of about 7m. The pits were filled with layers of radioactive waste, generally fixed with intermediate layers of concrete and/or sand, and then sealed with a concrete cap.

The overall mass and volume of the waste is estimated to be about 400 t and 243 m^3, with an activity of 45 TBq (mostly Ni-63, with only about 3 TBq of Co-60 and insignificant long lived radio-nuclides). Surveillance wells placed in proximity of the pits indicate no evidence of contamination of the underlying water table.

The recovery activities, based on a method already successfully applied in a validation test, will entail inserting a sleeve over each pit after excavating the surrounding ground, and immobilising the sleeve with cement to enable the pit to be safely lifted.

A method of (pre)treating and conditioning the Roman Pits is not yet defined, and three potential strategies are presently considered: a) boring out the original raw waste from the core of each pit, b) slicing each pit into segments or c) re-containerising each pit without any prior treatment.

A parametric study and a risk analysis of the three envisaged solutions has concluded that option a) should be the most robust and less risky one. However, until the design and the waste acceptance criteria of the final repository are fixed, it will not be possible to make a definitive choice.

The pits will be characterised during their temporary storage on site. Besides for confirming the radiological inventory data, the characterisation results will be used to assess the feasibility for any further treatment of the pits. Radiographic examination and DA of the pits cannot be excluded at this stage.

If no further treatment will be performed, prior to their shipment to the Italian National repository, the pits will be conditioned into special containers designed according to the repository requirements. It is envisaged that the containers will be essentially stainless steel liners with appropriate structural reinforcements for impact protection. The weight of each package will be approximately 40 t.

In case the alternative strategies will be applied the method of later undertaking intermediate and final characterisation measurements or sampling will be linked to the selected (pre-)treatment and conditioning processes.

Costs estimations are in favour of the solution requiring less treatment (9 M€ for option a) vs. 11 M€ for the two others).

The main project milestones foresee to start the pits extraction during the second half of 2011 and to complete their conditioning for final disposal by the end of 2016.

1) Unique Attributes/Challenges: Retrieval, characterisation and re-conditioning of waste disposed in large concrete package; characterisation of low β/γ emitters embedded in concrete; spatial constraints in the extraction area.
2) New Methodology Developments: TBD
3) New Equipment or Instrumentation Developments: Extraction and processing: none, nuclear use of conventional equipment; possible special portable characterisation equipment for Non Destructive Analysis.
4) Licensing and Political Issues: Qualification, licensing and acceptance by the Italian National repository of non standard packages; opinion of Safety Authority on the Ni-63 issue and on store in a near surface repository for LLW/ILW of solid waste classified as HLW.
5) Lessons Learned: From test extraction: vibrations during insertion of the steel sheet piles can influence neighbour structures; flooding of the pits excavation bay with the need to manage large quantities of soil and water (contamination check);

ILLW and sludge management

There are three main batches of ILLW. Two batches (~ 118 litres) are stored in two containers, “Cendrillons” type, in Area 40, while the third batch (~ 40 litres) is in an underground shielded tank in the former liquid effluents treatment facility (bd. 52). In addition to the above, there is a fourth batch constituted of a very small amount of ILLW (~ 20 litres) subdivided in three small containers stored in the hot cells facility.

It is believed that the “Cendrillons” contain dissolution of irradiated fuel rod with nitric acid with an estimated total α activity of 120 kBq/cm^3.

The ILLW stored in bd. 52 probably derives from the dissolution of irradiated fuel rods with nitric acid. The radiological analyses indicate an α-activity of 89 kBq/cm^3.

The ILLW stored in the hot cells facility has a measured contact exposure rate of 25 mGy/h.

If the results from analyses will show that the ILLW batches are chemically and radiological compatible with the sludge resulting from the past liquid effluents treatment, the least risk prone alternative would be to blend the streams and solidify the resulting mixture on-site.

Sludge on the Ispra site is present in area 52 and in about 200 drums (about 45 m^3). The tank in area 52 contains about 45 m^3 of concentrated effluent with α activity. The pre-characterised drums will be transported to the TFF. Once the sludge is stored in the TFF preliminary chemical analysis must be undertaken to develop appropriate recipe for the solidification. The sludge will be solidified in 440 l drums (CC-440 according to Italian UNI norms) using a proper cementation recipe. After the cementation, the packages will undergo the final dose and weight measurements, and will be labelled and transferred to the ISF.

Preliminary estimated costs, including final repository fees, in case the blending of the ILLW with sludge will be proven feasible should range from 7 to 10 M€. In case that the ILLW must be treated separately from sludge, the increase in costs will be conditioned by the possibility to treat the ILLW in an outside facility.

The main project milestones foresee to complete the ILLW recovery, chemical and radiological characterisation by the 1Q 2013, while by the 2nd half of 2012 all the sludge present in Ispra should have been transferred in the TFF, homogenised and characterised. Sludge and ILLW processing by cementation (if blending feasible) should be completed by 2017.

1) Unique Attributes/Challenges: ILLW retrieval and safety storage improvement; characterisation of waste starting from incomplete or poor historical records; processing ILLW possibly together with sludge; spatial constraints.
2) New Methodology Developments: ILLW/HLLW category downgrading by blending with sludge.
3) New Equipment or Instrumentation Developments: None
4) Licensing and Political Issues: Authorisation to ILLW/HLLW category downgrading and processing with sludge by cementation; qualification and licensing of the packages or authorisation to process them in Italy or abroad.
5) Lessons Learned: Past attempts showed that it is very difficult to implement external routes for ILLW treatment due to technical, licensing, and "political" reasons.

A2.1.4 Radiochemistry laboratory, basic nuclear facility 165 (formerly 57), France

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

Basic Nuclear Facility 57 consists of three buildings, Building 18 (plutonium chemical laboratory) and Buildings 91 and 54. Building 18 has an area of 8 000 m^2 (104×78 m). This building is divided into four fire zones making up units. Each unit has an area of 2 000 m^2 (40×50 m) and is composed of a hall (≤ 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 consisted 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.
- 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 more 98% complete on 31 December 2008. Progress with the individual tasks was as follows:

- Removal of nuclear materials: 100%
- Removal of radioactive sources: 100%
- Treatment and removal of aqueous effluents: 95%
- Treatment and removal of organic effluents: 100%
- Treatment and removal of waste: 95%
- Pumping out plutonium and transuranic contaminated solvent: 100%
- Flushing and decontamination of pond and pipes: 95%
- Cleaning of Buildings 18, 91 and 54: 95%

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.
- A treatment of very degraded Tri Lauryl Amine, by a alpha radiolysis (#1 000 W)
- The gel used for the decontamination of the glove boxes and the shielded lines (Aspigel 100 CeIV gel) was developed by the CEA [5]. This gel has the characteristic of drying by forming scales. Dried gel can be eliminated by vacuum cleaning or scraping.

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

A2.1.5 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 UF6 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. The decommissioning decree was obtained in February 2006 for duration of 5 years.

Decommissioning is to be undertaken in three phases:

- Final shutdown and post operational clean out (POCO): all reactants and uranium will be removed, dismantling of the process equipment.
- Process decommissioning: all process.

- General concrete clean up to allow non-radioactive waste mapping leading to the delicensing of the building and concrete clean up.

Due to a safety regulation change in the clean-up methodology, and technical problems with the subcontractors, the operations were stopped in October 2007.

April 2009 operations re started.

Due to these problems, an extension of the decree of 3 years is necessary in order to finish clean-up operations. The Delicensing (declassification of the facility) will be completed in 2015.

The total cost of the project is estimated at 60 MEuro (2010)

A2.1.6 ELAN IIB, France

Elan IIB was in operation (by CEA) at La Hague between 1970 and 1973 for the fabrication of 137Cs and 90Sr sources. It was shut down in 1973 for economic reasons. After emptying and rinsing process systems until 1977, it entered a dormancy period. In 1978, Cogema assumed responsibility of operations and carried out D&D operations between 1982 and 1991, after which it went into another dormancy period until 2000. Planning commenced in 2000 to restart post operations clean out. In 2004, ownership was given to AREVA and in 2005 ELAN IIB was integrated into the UP2 400 project. Project Management has been taken over by Cogema's team (ORCADE/UP2).

Post operations work continues under the operations venue while planning takes place and documentation is prepared for dismantling work. This work includes:

- Removal of brick barite walls between cells 900/813.
- Cells rinsing and cleaning.
- Installation of dismantling handling systems.
- Installation of waste handling, sorting and conditioning equipment.
- Removal of historic waste.
- Services upgrades.

The next proposed step is the removal of tank 1.21 from cell 900 to allow the investigation of the rest of the equipment and tanks in the cell.

Dismantling work can only be done after authorisation by decree. Six files are to be submitted including: Final state justification, Safety Report, Waste study, Environmental impact study and Emergency plan. Obtaining an authorization decree is planned for 2010.

A2.1.7 APM, Marcoule, France

APM (Atelier Pilote de Marcoule) is a facility owned by the Atomic Energy Commission and is situated within the Cogema plant site at Marcoule. APM was operated as an experimental facility for the treatment of irradiated fast breeder reactor fuel. The facility operated between 1962 and 1997, at which time reprocessing operations were terminated for obsolescence and economic reasons. This date (1997) also corresponds with the shutdown of the Cogema UP1 reprocessing plant where operations were directly linked to APM. APM operations also included the development of a vitrification process.

APM is comprised of three nuclear buildings:

- Building 211 was the main reprocessing building used to extract and purify uranium and plutonium, it also housed a variety of laboratories.
- Building 213 was used for the storage of vitrified waste, high level waste and radioactive sources.

- Building 214 received the spent fuel containers, where they were opened, sheared and dissolved.

The APM facility was finally shut down in June 1997 partly due to obsolescence of the facilities as well as for economic reasons.

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.

Although the reprocessing operations have been terminated, some areas of the facility remain in operation. The high-activity cells in Building 214 remain operational and are used for interim storage of highly active material, pending availability of disposition routes.

The basic decommissioning strategy is a phased approach, where an extensive decontamination programme will be followed by a dismantling campaign and cleanup of the buildings. The buildings will then be released from the nuclear envelope and will undergo conventional demolition at a later point in time. The decontamination phase will focus on minimizing surveillance, maintenance and operating costs; simplifying dismantling operations and producing only waste that can be surface-stored wherever possible.

The decontamination and rinsing of process systems has been ongoing for several years. A review of these operations and the early shutdown of AVM, which was used to process waste products, have resulted in a shift in strategy. Much of the process area is now planned to be dismantled by remote handling techniques, followed by decontamination of the waste.

The current schedule is for decontamination operations to continue through 2016, at which time the fuel will have been removed. Other HLW (alpha) will be dispositioned and MLW will have been transferred by this time. It is estimated that the dismantling safety licence will be obtained at the end of 2012 and this Phase will continue through 2020. The estimated budget is €300M for decontamination and €290M for dismantling.

Recent achievements include:

- Off-Gass Treament Unit - the removal of 12 glove boxes and associated piping has been accomplished (this task is complete).
- Planning for the dismantling and removal of the process water equipment, planning will be completed in 2009 and the Safety Approval process will continue throughout 2010.
- Planning for the removal of process equipment in the NUGG Cells which were used to store Cadarache NUGG fuels awaiting processing in UP1, the cleared space will be used to house a waste handling/processing facility.

A2.1.8 UP1, Marcoule, France

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

- Buildings 140, 144, 145, 146, 147 and 148 for reception, storage, shearing or decladding.

- Buildings 100 and 117 for dissolution, chemical extraction, separation, purification and plutonium production.
- Building 98 for uranium storage.
- Buildings 96 and 113 for liquids fission products storage.
- Building 130 for vitrification and wet storage.

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

UP1 was finally shutdown in December 1997 after treatment of all the French and Spanish GCR fuels (more than 18 000 tonnes 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 tonnes of process equipment.
  - 20 000 tonnes of structural material.
  - 6 shielded production lines.
  - 145 glove boxes.
- Very varied nuclear spectrum.
- Various radiological incidents during operation.
- Large quantity of operational waste.

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

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

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

The decommissioning operations are expected to take 25-30 years and cost MEuro 6 000 (including support facilities: laboratories, liquids and solids waste treatment).

The project has been separated in 3 units which concerned the decladding facility, the main factory and the vitrification plant. Work on these three projects has not progressed at the same level.

For the de-cladding facility, the major portion of the equipment has been removed. Waste in the pits has yet to be treated (Mg, Graphite, resins).

For the main plant, dismantling will require remote tooling which remains to be manufactured, in particular for dissolver tanks and evaporators in which activity is very high.

For the vitrification facility rinsing operations are not ended. Effluents stemming from rinsings of the fission product storage tanks will be glazed before this task is complete.

The work accomplished to date represents more than 2 million hours or half of the total forecast.

The main lessons learned are:

- “State of the art” process should be used for Characterisation, Cutting, Decontamination
- Equipment and methodology selection criteria are important for: Efficiency, Safety during implementation, By-products or effluents, Predictability.
- Some examples of specifically developed techniques: 3D gamma camera coupled to modeling; Laser cutting for remote handling; Gels and foams for decontamination.
- Remote handling successfully addressed used in difficult areas: Build. 117 HEPA filters room, Decladed fuel storage cell.

A2.1.9 SACLAY nuclear licensed facilities, France

Introduction: presentation and history

This CEA center was created at the beginning of the Fifties on the plateau of Saclay and is located approximately 25 km to the south of Paris. At the beginning of the year 2000 a cleansing and dismantling programme of the old Nuclear Licensed Facilities (NLF) was initiated. Currently a part of this programme relates to the Hot Laboratories (Laboratoires de Haute Activité: LHA), also called NLF 49 and the old workshops of the Liquid Waste Treatment Plant (Station des Effluents Liquides: STEL), also called NLF 35.

Other dismantling operations of the DEMSAC project (the decommissioning of the CEA-Saclay old Nuclear Facilities) concern:

- The dismantling of a laboratory of spent fuel examination (CELIMENE),
- The dismantling of old laboratory of “Service Hospitalier Frédéric Joliot”,
- The dismantling of buildings 156 and 196. The walls of these buildings are formed by concrete blocks and containing nuclear waste.

LHA dismantling programme

The LHA are multipurpose laboratories for R&D studies. It is an installation designed in the 1950’s and which had the role to accommodate the various CEA scientific services involving the production, and use of radionuclides. The first laboratories became functional in 1957 while parts of the Faciltiy were rebuilt in the 1980’s. The decision to shut down the laboratories was made in 1996.

The LHA are consisted by three distinct buildings.

- building 459; comprised of 16 Labs and a group of offices organized around a central corridor. Each laboratory contained an access air-lock with a joint office zone and technical rooms with a ventilation networks. The laboratories are separated each from another by a yard known as an inter-laboratory yard. In these yards are buried pits containing two laboratory liquid waste tanks,
- building 465 comprising 99Mo production shielded line,
- building 457 contains the NLF utilities and the general ventilation chimney.

One characteristic of the LHA dismantling project is the fact that the laboratories 4, 6 and 7 of building 459, the newest Labs, will be used for environmental protection studies, nuclear analysis labs and as a source storage facility in support of the decommissioning programme. It is necessary to physically isolate these facilities from the rest of the dismantling work.

End of LHA cleansing and preliminary dismantling

The End of the cleansing operations is defined by the removal of all irradiated material samples snf the cleansing of the shielded lines which had been dedicated to 137Cs sources manufacture.

Before the complete NLF dismantling the 9 LW tanks in the intra laboratory yards must be cleaned and removed. This operation is currently in progress. This work must be conducted to ensure containment of the hazards to the pits.

Dismantling scenario

The dismantling scenario selected comprises:

- the disassembling of the electromechanical equipment (shielded lines, the tanks and the collecting systems of RLW and the ventilation network);
- the civil-work structures cleansing;
- the civil-work demolition of structures, some laboratories, chimney and inter-laboratories yards.

The operations sequence follows a logic imposed by technical constraints, of safety and lawful procedures. It is articulated around 3 phases:

- Phase 1 is the dismantling lots presenting little project risk and little technical risk. However, it represents a large volume of working hours.
- Phase 2 will consist of a definition study of the characterisation methodology of the tritium contamination migration in the civil-work structures.
- Phase 3 will include the dismantling operations according to the methodology defined in phase 2.

STEL workshop dismantling programme

The NLF 35 or STEL (Saclay zone of radioactive liquid waste management – French acronym for Station de Traitement des Effluents Liquides), encompasses the collection, storage and processing equipments of aqueous liquid waste. Created in 1964, the NLF 35 underwent various process modifications (precipitation/filtration then evaporation/bitumen coating and today evaporation and concreting). The current evaporation process and the bitumen coating process was put into operation in the Seventies.

The cleansing and dismantling operations relate to the old processes of the STEL treatment (evaporation and concentrates coating: buildings 387) and the old storage tanks of the aqueous and organics MLLW and HLLW (building 393). These operations are carried out within the framework of the NLF 35 modification decree published in January 2004.

The objective of these operations is to dismantle and remove the process equipment in Building 387 and clean the rooms. In the case of Building 393, the objective is to remove access constraints to the building with the exception of those areas inherent in the operation of the NLF.

Building 387 processes dismantling

The dismantling operations of building 387 covers the tasks of the equipment removal, cutting and cleansing of the building rooms of the evaporation and coating workshops:

- the evaporator with its exchangers and receipt tanks;
- the bituminization systems and apparatus.

Taking into account the geometry of the buildings (the evaporator tower), the equipment will be disconnected then transferred into a dismantling workshop which can accommodate each piece of equipment.

Building 393 tanks cleansing

Before carrying out the cleansing and dismantling operations, it was necessary to realize:

- conformities of the electrical distribution, the radiation health physic monitoring and fire monitoring;
- modification of the ventilation chimney (E16);

- cleaning of the yard and to complete civil engineering works necessary for the reception of the LW tanks discharge in the lorry cistern.

To carry out the preliminary operations with the rising (LW chemical treatment so necessary and LW homogenization), draining and cleansing, an autonomous and mobile workshop will be set up. Two longitudinal beams are built on both sides of the tanks pit to be used as a track. The mobile workshop will move on these rails. The mobile workshop will also make it possible to carry out the tank dismantling.

The building 393 tanks are currently filled and will have to be drained:

- tanks MA 501 to MA 507 - old evaporation concentrates. They on the whole contain approximately 240 m3 of old concentrates to transfer by lorry cistern in the new workshop;
- tank MA 508 – organo-halogen LW. This tank contains approximately 6 m3. The pathway of this LW is CENTRACO;
- tank HA 4: 1,6 m3 of reprocessing HLLW organic and LWLW strongly contaminated in α. These effluents are treated by DELOS equipment (supercritical water OHT), established in ATALANTE facility in Marcoule CEA-Center. They will be transferred, after the discharge station installation, in a specific transport packing SORG, BU package for the organic solution transfers;
- tank VHA 3 - fission products residues (~50 litres).

Building 393 tanks dismantling

After obtaining an adequate residual contamination level (the decontamination target is: the dismantling waste will be LLW) all the processes and tanks will be dismantled and the sections evacuated as LLW waste:

- MA 501 to 507 tanks and MA 508 tank will be dismantled by using the mobile workshop. The tanks MA501 through MA507 will be dismantled progressively with their cleansing, the workshop goes tank by tank;
- pit 99 tanks. The tanks of pit 99 will be cleansed in order to allow their dismantling *in situ*;
- hall 1E cells MA tanks (4) were emptied at the end of the Seventies. It is a question of cleansing the pits (removal of waste, cartographies, identification and fixing the risks), of cleaning and of charting the externals of the tanks, to characterize the tanks interior then to dismantle them later on. Establishing an acceptable level of safety of the pits and the tanks is a precondition to the operations of tank HA 4 draining;
- hall 1E tanks HA 3 and HA 4. Once, tanks MA (see above) and tanks HA 3 and HA 4 cleansed, the hall 1E will be arranged to be able to evacuate the tanks for their dismantling in the cutting workshop;
- VHA process and its 3 tanks. Only tank VHA 3 was put in the active operation, it was emptied but was not cleansed. One of the first stages of the dismantling of this zone will be the characterisation of this tank.

The external pits will be demolished and the grounds handed over with the building 393 yard. On completion of cleansing of the the yard is then embanked and rehabilitated.

The building 393 halls will be rehabilitated in order to be possibly used for solid waste storage. The ventilation and the monitoring system will be adapted to this new activity.

New methodology developments

For the LHA decommissioning, studies of new characterisation technology for tritium contamination in the building structures are underway. Achievement of new techniques will hopefully reduce the costs of these buildings demolition. At present, the Nuclear Safety Authority requests buildings suspicious of tritium contamination be demolished after a erection of new ventilated enclosure.

New equipment developments

Aqueous LLLW and MLLW tanks of LHA and STEL are located on external areas of these facility buildings. For the cleansing and the dismantling of these tanks it was necessary to study and carried out ventilated workshops. These workshops replenished containment barriers in regard of the regulatory compliance.

To move from tank to tank, according to the provisions of the areas and vessels, these workshops moves on rails for the STEL tanks or is moved by crane handling for the LHA tanks.

To inspect tanks, to characterize deposits, and to set the contamination, it has developed a mobile Gloves Box. This GB allows you to perform these operations before the establishment of ventilated workshops, enabling a better time schedule control and reduce project risks.

Licensing and polical issues

LHA dismantling

Licensing/time schedule and waste production

The MAD/DEM decree was published on September 18, 2008. The decommissioning must be completed in 2018. The LHA dismantling global planning is:

- 2000-2009: cleansing and dismantling preparation (safety files, Industrial arrangement procedure),
- 2009-2013: phase 1 of the dismantling works (VLLW # 1 800 Mg, LLW # 150 m^3, HLW # 0.5 m^3) and the phase 2 (studies),
- 2014-2017: phase 3 of the dismantling works (VLLW # 7, 000 Mg).
- The total cost is evaluated to 83 M€. The final state is the reusing of the building (0.4 Bq/cm^2 in β/γ nuclides). Just the old laboratories for tritium R&D are demolished.

Industrial arrangement

The industrial arrangement for LHA dismantling phase 1 rests on a total subcontracting including the services:

- dismantling works;
- facility operational exploitation;
- the waste management (production flow development, the conditioning, the characterisation, the preparation of the service request files, the intermediate storage of the waste packages with the storage management, and the controls before evacuation).

STEL old wokshop dismantling

The cleansing and dismantling operations relate to the old processes of the STEL treatment (evaporation and concentrates coating: buildings 387) and the old storage tanks of the aqueous and organics Medium and Height LLW (building 393). These operations are carried out within the framework of the NLF 35 modification decree published in January 2004. The total cost is evaluated to 28 M€.

The STEL dismantling global planning is:

- 2004-2009: preparation [safety files, general studies and preparations works cleaning of the yard (VLLW # 10 Mg, LLW # 450 m^3)];
- 2010-2014: building 387 dismantling works (VLLW # 1,500 Mg, LLW # 200 m^3);
- 2010-2016: building 393 dismantling works (VLLW # 25 Mg, LLW # 100 m^3).

The final state is the reusing of the building (0.4 Bq/cm^2 in β/γ nuclides and 0.04 Bq/cm^2 in α nuclide).

Interests of the project, conclusions and discussions

The DEMSAC project has several points of interest:

- For the LHA dismantling:
  - It is the first global operation entrusted to a general company including preliminary studies, dismantling works and the facility technical operation,
  - Some laboratories are contaminated by tritium. As part of the project, a R&D programme for the Tritium in-situ measurement has been initiated. The target is the cost reduction of the final decommissioning,
- For the STEL dismantling, it is a significant dismantling operation in a NLF in operation.

These operations correspond to the first operations of CEA-SACLAY Center NLF dismantling. They gradually began during year 2000 and will be continue until the end of the years 2010.

For the LHA dismantling some of the main difficulties arise from the poor state of documentation in the old facility.

A2.1.10 Wiederaufarbeitungsanlage Karlsruhe (WAK), Germany

The Karlsruhe Reprocessing Plant (WAK) was shut down in 1991. The facility consisted of the main process building and HWL (the old HLLW Facility). A new High HLLW Facility (LAVA) has been established and a vitrification facility (VEK) has been constructed and is being commissioned to process the historical HLLW. The original intent was to restore the site to green-field state by about 2014.

The contract for completion of the project has been turned over to EWN who now have complete technical and financial responsibility for the project. The current schedule is to complete the vitrification process in 2010 and dismantle and release the HLLW facilities (HWL, LAVA, VEK) by 2020 and finally to demolish the buildings (green-field) by 2023.

(WAK) was a pilot reprocessing facility located on the grounds of the research centre of Karlsruhe (FZK). It was shut down in 1991, 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.

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

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

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

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

Dismantling the WAK started in 1993. To date the following has been accomplished:

Main process building

- The HLLW facilities have been technically separated.
- 2 000 tonnes of equipment has been dismantled and 1 100 tonnes of debris disposed of 5 500 m^2 of wall surfaces (of 6 600 m2) have been shaved.
- 96% of the radioactive inventory (4.8E14 Bq) has been transferred to interim storage.
- The programme for radiological measurement continues.

HLLW facilities

- A new access building to HWL, and LAVA has been in operation since May 2008.
- The LAVA HLLW inventory (60 m3), included 500 Kg of Uranium and 16.5 Kg of Plutonium.
- HWL is out of operation and will be used for spare storage.
- All preparations for the remote dismantling of the MLLW tanks are complete and the dismantling of five tanks is underway.
- An exterior and interior video inspection of a HLLW tank and sampling of residue is complete, analysis has revealed no major surprises.
- The licence for dismantling the HLLW tanks is still being assessed by the Authorities.

Vitrification facility (VEK)

- Cold tests are completed (June 2009).
- VEK connected to the liquid waste storage facilities (LAVA), including transfer lines and off-gas systems.
- Operating licence granted (February 2009).
- Hot tests completed (September 2009).
- Hot operation started (September 2009).
- The vitrification process was completed in June 2010, after which a rinsing operation will begin.

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.

An interesting comparison could be made between the manual dismantling of cell VII and the vertical remote dismantling of cell V. In cell VII, 4 800 man-hours were used to dismantle 17 t of material with a total activity of 5×1 011 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 V, 18 000 man-hours produced 33 t of material with 3×1011 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.

A2.1.11 SOGIN – pilot U-Th reprocessing plant

The ITREC Plant, a pilot scale plant for Thorium-Uranium Fuel Cycle research, is located at the Trisaia Site on the South East Coast of Italy and owned by SOGIN. The plant is made up of three main components; the main process building (which includes a 60 meter stack), the liquid waste storage areas (twelve underground storage tanks in two buildings – Waste 1 and Waste 2) and the solid waste warehouses. The plant construction started in 1960 within the framework of the co-operative (US/Italy) U-Thorium Fuel Cycle Programme. Construction was completed in 1970, followed by five years of non-nuclear tests and three years of nuclear tests.

The plant was originally conceived as an integrated unit for reprocessing and remote re-fabrication of fuel elements. The hot cells; where fuel cutting, dissolving and solvent extraction took place, were built up of modular units that could be removed by remote controlled operation for decontamination, maintenance and modification. This feature will facilitate the decommissioning process. The warm cells where the fuel re-fabrication was to take place were also remotely operated.

From 1968 to 1974, non nuclear tests were carried out in the cells with natural Uranium-Thorium. These tests ended after the responsible government department terminated the fuel re-fabrication programme and the warm cells were dismantled. From 1975 to 1978, 20 ERR fuel elements were processed in the hot cells. Solid wastes were transferred to above ground storage facilities while the liquid wastes (50.7 m^3/3 01.5 TBq) and U-Th liquor (3.2 m^3/4.0MBq) were transferred to underground storage tanks. From 1979 to 1990, the facility was used for component and process development.

Waste management

In 1995/1996, the first cementation programme of LLW took place. The SIRTE-MOWA facility processed 80 m^3 of LLW, resulting in 443 cemented containers. In 1999, the second cementation campaign took place whereby 3m^3 of HLW were mixed with 47 m^3 of contaminated cleaning solution, resulting in 337 cemented containers. From 1989 through 2006, 7 500 drums of solid LLW were reduced (compacted) to 1 181 re-packaged drums in three separate campaigns. From 2001 through 2007, 5 348 empty oil drums were decontaminated; most were sent to a conventional foundry in Italy for melting. In 1994, 1 200m^3 of slightly contaminated soil was characterized.

As of July 2006, IRTEC was issued an operating license to maintain safety at the facility and to continue preliminary work in advance of final decommissioning. The license obligates SOGIN to file the final decommissioning documentation by July 25th, 2011. The planning process is underway.

Activities that will be carried out under the operating license include:

- Dry storage maintenance of 64 ERR fuel elements (ongoing).
- Solidification of the Thorium/Uranium Liquor (ongoing).
- Removal and remediation of HLSW pit (ongoing).
- HLSW and LLSW treatment (ongoing).
- Work applicable to global decommissioning planning (characterisation).

Unique attributes/challenges

- Type of fuel: Irradiated Thorium/HEU MOX fuel.
- Type of reprocessing: Thorex process.
- Some minor accidents during operation may impact decommissioning (contamination of some soil and containment pipe/liners).
- There are very small amount (three cubic meter) of nitric acid solution from reprocessing of peculiar spent exotic fuel.
- New information could came from this project regards conditioning of high alpha contaminated and gamma emitter liquid waste by means of cementation.

New methodology developments

- None at the moment.

New equipment or instrumentation developments

- Dedicated cementation mock-up pilot plant.

Licensing and political issues

- Uncertainties for several years in nuclear policy at central government in Rome had as consequences more high NIMBY or a priori opposition to nuclear activities at local level.
- Several revision of the long period view reflects on unreliable timetable and cost forecasting in the short/middle time activities.

Lessons learned

- More effective communication and information on nuclear risk would speed nuclear activities as they would be seen as useful more than dangerous.
- Clear view on long term period should be reliable.

A2.1.12 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 grammes 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.
- Non-purified 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 non-purified 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 absorbers. The treatment of all liquid wastes was completed in 1998.

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

The research and development programme resulted in the following:

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

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

The glove boxes were dismantled in 1996 to prepare the space for temporary waste storage yard. Then, the analytical cells which consists of an inner box of stainless steel with iron and lead shielding, components in hot cave, solvent recovery cell, Pu cell and large sized tanks were dismantled. As a result, about 50% 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.

Detail information during these 5-years

Unique attributes/challenges

- None.

New methodology developments

- Wall-penetrating pipes in reprocessing facility were curved in walls and it was hard to separate the pipes from the concrete walls. We removed the pipes from the facility with concrete walls around them, using dry coring system and dry wire-saw not to spread radioactive contamination in the facility. The removed concrete was broken by static demolition agent to separate the pipes from concrete. It was confirmed that this method was effective for removing curved wall-penetrating pipe in reprocessing facilities.
- To dismantle liquid waste storage tanks placed in narrow concrete cell, the tank was removed from the cell without dismantling, transferred to a dismantling facility, and dismantled there. It was confirmed that this method reduced radiation exposure, waste generation, and various risks in dismantling, compared to methods of dismantling the tank in a narrow cell.

New equipment or instrumentation developments

- To dismantle liquid waste storage tank containing strontium insoluble residue, radiation exposure of workers should be reduced. We made a ventilated suit covered with vinyl chloride to shield β-ray emitted from strontium and the dose rate inside and outside the suit were measured. It was confirmed that the β-ray was shielded and exposure of worker could be reduced to a large extent by the suit.

A2.1.13 Uranium refining/conversion facility

The Japan Atomic Energy Agency (JAEA) has a Uranium Refining/Conversion Facility at the Ningyo-Toge Environmental Engineering Centre near Osaka Japan. The facility was constructed in the 1960s after the discovery of uranium in 1955 and operated until 2000. The site is large, encompassing approximately 1.2 Million square meters and hosting a total of 31 nuclear fuel facilities. The main facilities are the Refining and Conversion Plant, the Enrichment Plant and the Milling Facility. This project includes the D&D of the Refining and Conversion Plant and the Uranium Enrichment Demonstration Plant.

The Refining and Conversion Plant was initially used for the examination of natural uranium and lately for reprocessed uranium. The building has a multi-level structure containing the complex processing systems. A detailed description of the conversion process, the process equipment and the containment building are given in the presentation material on the DVD.

The Uranium Enrichment Demonstration Plant operated until 2001, performing enrichment of both natural and reprocessed uranium. After shutdown, decontamination of the facility had been carried out using IF7 gas.

The general objectives of the decommissioning plan are:

- To develop techniques and processes that minimize the cost of decontamination, dismantling and waste disposal
- To minimize the volume of waste, both primary and secondary
- To complete the project on a short time scale (prompt decommissioning)
- To decontaminate and recycle as much metal as possible
- To involve the local community and have an overall positive benefit
- To apply the experience gained to other projects

Dismantling of the Refining and Conversion Plant was begun in 2008. Technical demonstrations will be conducted in the Uranium Enrichment Plant until 2011, after which dismantling will be started.

Unique attributes/challenges

Uranium refining and conversion facility

The Uranium Refining and Conversion Plant Facility (URCP) converted about 340 tonnes of recovered uranium from reprocessing plant. Amount of remnant (contaminated) uranium in the plant is small. Therefore, special radiation protection and remote dismantling technology are unnecessary for URCP dismantling.

Liquids (UCl_4, UO_3+H_2O), fine particles (UO_3, UF_4) and gases (F_2, UF6) are used at URCP. Therefore, equipment contamination is different by process. URCP dismantling needs following careful consideration.

- Liquid process: prevention measures of secondary contamination from uranium liquid leakage.
- Fine particle process: prevention measures of uranium containing fine particle dust from equipment cutting.
- Gas process: prevention measures of Hydrogen fluoride generating by reaction of Moisture and uranium fluoride.(UF_6 ,UF_5,UnFm)

About 1 300 tonnes of uranium sludge [ex., neutralization sludge (CaF_2, $CaSO_4$)] and under dry conditions contaminated materials (ex., uranium fluoride compound absorbent (NaF, Al_2O_3, MgF_2, Na2CO3), fluidization media (γ-alumina) and diatom earth (SiO_2) have been generated from the URCP operation. The uranium content of these types of waste is relatively high. Therefore, it is necessary to decontaminate the uranium to dispose these wastes.

These wastes are stored in drums and kept in facilities except fluidization media which is kept in tanks and buried under ground. In our plan, evacuating fluidization media comes first, and decontamination of inner surfaces of the tank are followed. The tanks will be dismantled at the final stage.

In addition, the uranium may have infiltrated into the floor face concrete. Therefore, measurement and decontamination of those are necessary.

Uranium enrichment demonstration plant

For the decommissioning of the Uranium Enrichment Demonstration Plant (UEDP), no particular radiological protection measures and dismantling technology such as remote dismantling technology is required.

According to Nuclear Non-Proliferation Treaty, however, management of sensitive information on uranium enrichment is required.

Therefore, during dismantling of the facilities, it is important to complete termination processing of sensitive information simultaneously.

Concrete measures for the termination processing is under discussion at present.

About 3 tonnes of adherent uranium fluoride compound are inside of the UEDP. This uranium fluoride compound is to be recovered by system decontamination before UEDP dismantling.

Dismantling data acquisition

We use a mark sensing card for collecting the dismantling result data. Collected data is used for the dismantling cost evaluation code (COSMARD). And text database are made from experience or lessons learned information from the dismantling.

New methodology developments

Chemical reaction decontamination technique using IF_7 gas

The chemical reaction decontamination technique using IF_7 gas developed as a system decontamination technology for the vacuum plant like UEDP. This technique has an advantage of generating less secondary waste.

Decontamination of uranium containing sludge and granular materials

We are developing three types of decontamination technologies for these wastes. These technologies will be combined depending on features of the uranium containing sludge.

- Specific gravity separation by dry method.
- Solvent extraction by emulsion flow.
- Coagulating sedimentation by hydrochloric acid dissolution.

New equipment or instrumentation developments

In the URCP and UEDP, the cutting work is done with a mainly with a mechanical saw. Reliability and useful life of the mechanical saw are important for an efficient dismantling work. Therefore, we use new type of diamond cutter. This is for both metal and concrete.

Licensing and political issues

- Decommissioning license approval requests of transferring about 2 300 tonnes of depleted uranium which are kept in UEDP. However, this transferring is very difficult task.
- It is not possible to dispose of uranium radioactive waste in Japan since related legislation has not been prepared yet. Clearance metal is recyclable under the Japanese law for nuclear facilities use only.
- We have not determined yet how to deal with the building of URCP and UEDP lastly (dismantling or another purpose use).

Lessons learned

The URCP dismantling began in 2008 and we have limited experienced to date.

From our view, the lump-sum contract cost too much. The issue is management costs consist of administrative cost and hotel and transportation cost. Therefore, we changed contract from lump sum major company contract to local company temporary staffing contract from FY2009 for cost reduction.

A2.1.14 Plutonium Fuel Fabrication Facility (PFFF), Japan

Plutonium Fuel Development Center of Japan Atomic Energy Agency (JAEA) has 3 facilities for the development of plutonium fuel:

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

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

- Phase 1 (up to 2015): stabilisation and ready for shipment of nuclear material in the facility. Pelletizing and assembling will be chosen.
- Phase 2 (2005-2015): D&D planning and adaptability tests.
- Phase 3 (2010-2030): size reduction of equipment and GB.
- Phase 4 (2010-): re-use of buildings for waste storage.

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

Unique attributes/challenges

We size reduce combustible and PVC included waste markedly by incinerating them.

New methodology developments

A visual support system using laser rangefinder (re-create information of work space) to help remote operation has been developed.

New equipment or Instrument developments:

In support of secondary waste and mixed waste reduction, we are testing a double cover type drum into which we can directly put the size reduced waste. We incinerate combustible and PVC included waste in Plutonium Waste Treatment Facility.

Licensing and political issues

Storage of generated waste is current issue.

Lessons learned

Still need more experiences to draw the lessons learned.

A2.1.15 Uranium Conversion Plant, Korea

The construction of the uranium conversion plant (UCP) of KAERI (Korea Atomic Energy Research Institute), with a capacity of converting 100 t/year of uranium concentrate to UO2, was commenced in 1976 with a purpose of the localization of the nuclear fuel manufacturing technologies. Its operation was started in 1982 and the process technologies were improved to add an in-house developed process, the AUC (ammonium uranyl carbonate) process in 1987. It was also used for producing 320 t UO2 for fuel for the Wolsong-1 CANDU reactor. The plant was shut down in 1992, due to the economical unfeasibility from the small capacity of the plant and reshuffling policy of nuclear industries of the Korean Government. It was decided that the plant would be dismantled in 2000 and a project for decommissioning the plant was launched in 2001 with the goal of a completion by 2007. The approved budget was 91.5 MUSD and the scopes of the project were the dismantlement of all equipment and removal of the entire radioactivity to use the building as a non-nuclear purpose.

2001 ~ 2004; preparative period for the project. A plan for the decommissioning project of the KRR-1 and 2 was prepared by KAERI with a help of the experience from the KRR decommissioning. In this project, the same strategies as the KRR decommissioning project were selected; an immediate dismantling, minimum radioactive waste generation and simultaneous technology development. A decommissioning plan including the EIA report and the QAP (quality assurance programme) was submitted to the Ministry of Education, Science and Technology (MEST) for the license in June 2006 and was approved in August 2004 after a safety review.

2005~2006; dismantling period. All the equipment, chemical reactors, tanks, pump and all accessories, were dismantled and cut into small enough pieces to feed into the decontamination process. The dismantling works were carried out room by room, considering working space, waste discharge route and prevention of re-contamination by cutting works.

2007~ 2010; decontamination and waste treatment period. The metal pieces from tanks, vessels, pipes, structures and so on were decontaminated by two methods; stainless steel by ultrasonic cleaner and carbon steel by induction melting. By the decontamination works, more than 70% of metal waste is expected to release to general steel industries. The inner wall and the floor of the building were decontaminated with grinders, in-house fabricated wall shaving machine and small excavators. The lagoon sludge waste was treated to the stable powder for long storage awaiting a decision for the disposal of the uranium contaminated waste. Because of the high manpower requirements, the target date of the project was delayed to 2010.

Unique attributes and/or challenges

During the decontamination of the floor of the building after dismantling all equipment in the plant, it was found that the soil under the building was contaminated with uranium compounds by a penetration through the cracks of the concrete. The depth of the contaminated soil was up to 6 meters. It was concluded that the contaminated soil would be removed and the building would be modified and reused as a laboratory where uranium will be handled. But the 6 meters is deeper than the feet of the building base and therefore reinforcing structures will have to be installed during removal of the soil for maintaining the structural integrity of the building. More than 600 m3 of the contaminated soil waste is expected. In order to reduce the volume of the soil waste, a soil decontamination process is being developed and the soil is temporary stored in the lagoons after some modification.

New methodology development

In the UCP, there were two lagoons, where all the liquid waste generated from the operation of the plant was stored. During a 10 years shut down period, the water was evaporated to produce a sludge waste. The main components of the sludge were nitrate salts of sodium, ammonium and calcium and the sludge also contained uranium and several heavy metals. It was a mixture of chemical and radiological hazards. The treatment process was developed through a joint R&D campaign with Cadarache Research Center of CEA, France. The candidate processes were thermal disposition process and biological de-nitration process. Finally the thermal decomposition process was selected and a facility was installed to treat the sludge of 250 tonnes. The process consisted of pre-treatment, thermal heating, powder packing and off gas treatment. A two step heating process was adopted for solving the difficulty of large volume expansion (about 1000 times) of the sludge during heating.

For the purification of the uranium for nuclear fuel manufacturing, a mixture of TBP and dodecane was used as an organic solvent. About 4 m3 of this uranium bearing organic solvent remained in the equipment of the plant. It was forbidden to incinerate this material due to the possible spread of uranium to the environments. A hydrolysis process was installed and the TBP was decomposed to butanol in alkali solution. The dodecane and the butanol were separated by distillation in a reduced pressure. All the uranium contaminates remained in aqueous alkaline solution and the not-contaminated organic materials were recovered and will be incinerated.

New equipment and instrumentation

For the dismantling of the UCP, all the equipment and instrument were available in open markets.

Licensing

None.

Lessons learned

The final target date of the project was the end of 2010 and the final condition after decommissioning was a free release of the buildings for non-nuclear purpose. During decommissioning project, the final conditions were changed to the conditional release and the building should be reused as a uranium handling laboratory from 2010. The target date was rearranged to 2009. When UCP was characterized, the contamination of the soil under the floor of the building was not detected as the trenches under the vessels and tanks were lined with stainless steel plate. But it was found near final stage of the building decontamination that large amount of the soil was contaminated. The contaminated soil should be removed without demolition of the building in order to reuse it. Because of two changed conditions, the decommissioning plan was reassessed and the target date of the project was delayed again to 2010 and the the budget will be increased by about 25%. The treatment and/or disposal of the contaminated soils are not included in the budget of completion date.

A2.1.16 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 approximately 30.5m x 18m in plan and approximately 61 m in height surmounted by a 61 m high ventilation stack. The two original mirror image process lines each comprise two highly active cells and a medium active cell. The cells were open for their full height with process vessels and pipes supported on mild steel joists.

At an early stage the project was divided into nine phases, covering a period of approximately 20 years. The following progress has been made against the original 9 phases:

Phase 1 - complete	A test-bed Waste Handling Facility was designed and installed in the 1990s, but experienced significant difficulties throughout its inactive commissioning stage and was never made active. The facility is now obsolete and a new facility will be required in the base of the MAN cell as part of the revised decommissioning strategy.
Phase 2 - complete	Conceptual design studies for high active and remaining medium active cells have been completed. Additional data has been obtained using laser scanning and N-Visage dose surveying/modelling for the high and medium active cells, although further data is still required for some key areas.
Phase 3 - complete	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.
Phase 4	Decommissioning the MAN cell started with the removal of the associated control systems, out-cell services and vessels/pipes from 5th floor upwards. 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. HANO stabilisation is complete for the main shaft and the upper cell areas, above the MAN cell, will be completed in 2010.
Phase 5 - complete	Emptying the stainless-steel hulls silo was completed in 2000. A flask loading facility was installed, with a remotely operated loading vehicle (ROV). The ambient dose rates in the silo were up to 200mSv/h gamma. Despite initial reliability problems with the Remotely Operated Vehicle (ROV) the work was successfully completed, reducing general dose rates to 10mSv/h gamma.
Phase 6	Decommissioning of the Medium Active South (MAS) cell has not started.
Phase 7 and 8	Decommissioning of the high active cells north and south outer (HANO/HASO) is ongoing. HANO was used 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 vessel and pipework support steelwork. 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 main shaft section of the cell has been successfully stabilised with a bespoke light-weight foam grout formula. Work is currently ongoing to complete the stabilisation of the upper cell area in 2010. HASO cell decommissioning has progressed significantly, with manual removal of vessels and pipes from the 8th floor down to the 5th floor. Work will continue down to the 2nd floor using manual techniques, but below this level higher levels of radiation will require a less hands-on approach. HANI/HASI cells, Filter Silo, Head End Plant, Ejector Room and Sampling Laboratory have not yet commenced. Various out-cell clearance activities have been undertaken in a fortuitous manner, including removal of 12 Instrument Bulges and removal of redundant control panels, pipes and cables from annulus areas.
Phase 9	Removal of original Ventilation systems, Effluent systems and Final Clearance of the building has not yet commenced.

Following seismic assessment of the main building structure and the ventilation stack, 2 key projects were established around 2005 to remove the ventilation stack, and to provide alternative ventilation discharge via a new stack. This work is planned for completion in 2015.

The original 9-phase decommissioning strategy has been significantly revised in 2009 as a result of difficulties associated with removal of the cell vessels. The revised decommissioning strategy takes advantage of the current projects to remove the stack, making it possible to set up overhead travelling cranes above the HA and MA cells, with a single waste handling facility at the ground floor level in the MAN cell, as shown in Figure 2. Thus, the transfer of plant items from the cells becomes a simpler vertical lifting operation, controlled from outside the cells. Plant isolations, and rigging for transfer will still need to be done remotely, as will the packaging of the waste.

The revised strategy also simplifies the handling and treatment processes that would be carried out actually within B204 by maximizing use of the Sellafield Site's proposed Decommissioning ILW Encapsulation Plant (DILWEP) and PCM Waste Handling Plant, which will have the capability to open up various waste packages, size reduce and repackage the contents ready for long term storage. The use of the above plants will allow larger packages of waste to be dispatched from B204, with less work carried out in the building.

The revised strategy also includes a more accurate waste inventory assessment and a re-assessment of the cost and schedule to complete the Project. The total project cost, inclusive of surveillance and maintenance costs, at P80 confidence are £570M with a forecast completion date of 2090. The extended duration is primarily associated with site funding constraints and links to the construction of new waste handling facilities.

Unique attributes/challenges

- 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. Some of the modifications were not fully implemented leaving an ambiguous status for some cell areas, particularly regarding the scale and completion of process wash-outs.
- A series of spillages have contaminated in cell and out cell areas of the process cell structure.
- Severe corrosion of the mild-steel support steelwork in the HANO cell has prevented some decommissioning activities.
- Low seismic with stand of the building and stack are driving the decommissioning strategy. Removal of the stack is now required well in advance of main building structure demolition.

New methodology developments

- Utilisation of laser scanning technology for high-radiation cells and areas where personnel access if difficult (Shear Cave, HANI & HANO).
- Development of bespoke light-weight foaming grout at a density of 300 to 500 kg/m3 to stabilize the vessels and pipes within the HANO cell.
- Use of a fogging decontamination agent to reduce airborne contamination
- Use of radiosensitive paints
- Development of Plasma Arc cutting techniques

New equipment or instrumentation developments

- Utlilisation of a novel radiation survey and modeling system (N-Visage) to assist in characterisation and decommissioning planning of Shear Cave and HASO cells (Figure 3). It is currently being applied to HANO and HANI cells.

Licensing and political issues

- The baseline safety case for the building has been categorized A due to external hazards assessments with potential for impact on adjacent plants.

Lessons learned

- Early decommissioning strategy modified as a result of changing drivers.
- Movement away from remote handling to more hands-on approach wherever possible due to early development and operation difficulties.
- The current Waste Handling Facility (WHF) has proved through its relatively low level of success during the postponed MAN clearance project that it will need to be replaced entirely. The location of this facility is also questionable with the preferred location being identified as the ground floor of the MAN Cell.

A2.1.17 B243 Intermediate Waste Recovery (BNG/Sellafield)

B243 was constructed in 1951, as part of the original suite of plants for the management of low active effluent arising from the Sellafield site. The facility consisted of eight cells which house sand bed filters, used to remove active particulate from the effluent stream prior to sea discharge. However, the sand beds were never used and were subsequently removed. It was decided that the empty shielded cells provided a suitable facility for the storage of beta/gamma waste unsuitable for disposal at the LLW repository. B243 received beta/gamma waste into the shielded cells from 1970 until 1986.

In 1989, an overbuilding was constructed over the six northern cells (cells

1 to 3 and 6 to 8). Cells 4 and 5 were subsequently clad in 1999. In addition to construction of the overbuilding, the facility was enhanced to allow storage of beta/gamma waste in the areas between the cells. This work included the provision of shield walls and associated foundations between cells, in-filling valve chambers and service ducts between cells and the installation of a new working area floor slab.

The shielded area around the cells was used from 1990 until 1996 for the receipt and storage of beta-gamma waste in the form of LLW and ILW filter boxes and 200 liter drums. No further waste has been received into B243 since 1996.

In 2000, BNFL were issued with a License Instrument Specification in relation to B243. The specification was issued as a result of executive concerns regarding the substantial amounts of ILW in a raw form, stored in a facility not intended for long storage and the potentially mobile state of contamination due to water infiltration. The License Instrument states that the Licensee should not accumulate radioactive waste except in a manner and place approved by the executive; this specification comes into effect August 01, 2010. This imposes a requirement to condition the waste in the cells for long term passive storage.

A programme of work (the B243 decommissioning project) to fulfill the requirements of the License Instrument Specification is currently in delivery and will install an overbuilding with environmental monitoring and fire detection system to act as a suitable temporary storage facility. The facility will also include mechanical handling and containment systems to enable retrieval of the original cell covers and characterisation of the cell contents. Based on the data obtained a decision will be made in 2013/14 regarding the waste recovery methods and priority.

On current schedule, the overbuilding will be commissioned by July 2010, followed by full commissioning of the characterisation systems by September 2010 at a cost of approximately £23.5 million UK pounds. The project phase will then be complete and the facility will move into an operations phase for characterisation and recovery of the waste.

The total project and operations cost associated with removal of the facility and inventory is estimated at £34.1 million UK pounds and building demolition is scheduled for 2040.

Unique attributes/challenges

- Miscellaneous, uncharacterized wastes stored with poor inventory record.
- Spatial constraints surrounding the plant make construction and decommissioning activities very difficult

New methodology developments

- None – as yet looking at characterisation techniques to assist sort and segregation of the wastes where practicable.

New equipment or instrumentation developments

- None

Licensing and political issues

- License Specification issued as a result of Executive concerns regarding the substantial amounts of ILW in a raw form, stored in a facility not intended for long storage and the potentially mobile state of contamination due to water infiltration. See above for details.

Lessons learned

- The original plan to decommission the cells involved the construction of a complex remote handling facility to deal with the contents of the cells. Following a strategic review, it was identified that there is potential for some quantities of waste to be retrieved using manual and/or semi-remote techniques and sentencing as LLW rather than ILW. The project therefore decided to limit the design and installation of complex equipment pending further characterisation of the cell waste inventory.

- The Site Licensee initially misunderstood the requirements of the License Specification and it was only following detailed discussions with the Regulator that a common understanding occurred.

A2.1.18 Portsmouth Gaseous Diffusion Plant (Portsmouth D&D)

Organisation:	U.S. Department of Energy (DOE)
Type of Facility:	Uranium Enrichment: High Enrichment Uranium (HEU) for national defense, Low Enrichment Uranium (LEU) for commercial uses.
Brief Description:	DOE-owned, contractor-managed, 3 777 acre site; the Portsmouth gaseous diffusion plant (GDP) area is 1 200 acres; located: Pike County, Ohio, built 1952-1956; uranium processing starts 1954. DOE, and predecessor agencies operated GDP; GDP leased to US Enrichment Corp. (USEC) and operated by USEC under certificate from Nuclear Regulatory Commission. USEC ended enrichment 2001. Site in Cold Shutdown. DOE issued Request for Proposals for GDP Decontamination and Decommissioning (D&D); bid evaluations ongoing; contract award expected 4th quarter 2010. Number of employees: 2 700.

Operational details

Operational Period

- Uranium Enrichment: 1954 to 2001.
- Peak Capacity: 7.9 Million Separative Work Units (SWU)/year.
- Total throughput: > 300 Million Kg Uranium/450 Million Kg UF6.

History

US built three GDPs for national weapons programme and commercial nuclear reactor fuel. GDP facilities also built at Oak Ridge, Tennessee, and Paducah, Kentucky. Portsmouth uranium enrichment consisted of 4 000 stage GDP cascade located in three large Process Buildings with 96 acre collective footprint/roof area. Extensive supporting infrastructure and utilities also built.

DOE leased GDP facilities to USEC 1993. USEC produced Low Enriched Uranium for commercial use from 1993 to 2000. USEC ended enrichment at Portsmouth in 2001. DOE placed site in Cold Stand-by in 2001 and Cold Shutdown Programme in 2005.

DOE built Gas Centrifuge Enrichment Plant (GCEP) on site in 1980s, completed two process buildings. Centrifuge machines installed; project terminated 1985.

DOE built depleted uranium conversion facility in 2008. Operations to convert depleted uranium hexafluoride to an oxide are scheduled to begin in 2010.

USEC has leased the former DOE centrifuge facilities and is currently installing and testing new centrifuge equipment for commercial enrichment production, pending the approval of financing for the project.

Portsmouth Site Tenants

Other public and private tenants at Portsmouth, and their missions, are:

- Uranium Disposition Services, LLC, to design, build, operate system to treat, and dispose site's inventory of depleted uranium hexafluoride (DUF6).
- Wastren-EnergX Mission Support, LLC, Facility Site Services contract.

- LATA/Parallax, Portsmouth, LLC, Remediation contract.
- USEC, Corp. Cold Shutdown and site services contract.
- USEC Inc., commercial business, American Centrifuge Plant (ACP).
- U.S. Army/Ohio National Guard; leases heavy equipment garage and storage space.

Portsmouth Remediation Challenges

Management and remediation of site presents the following challenges:

- Avoid releases of hazardous/objectionable materials during remediation.
- Transition work force to D&D; utilize their site training and knowledge.
- Remediate 5 trichloroethylene (TCE) contamination plumes and point sources.
- Complete pre-D&D preparation work in Process Buildings and other facilities to reduce hazards for D&D.
- Evaluate site utilities to maintain services to site tenants.
- Interface with regulatory agencies and stakeholders to define remediation requirements.
- Interface with stakeholders to obtain input on preferred future land uses.

Basic decommissioning strategy

Site buildings/facilities decommissioning work scope will:

1. Accept facilities as they are de-leased and returned to DOE by USEC.
2. Characterize and determine regulatory D&D and waste/recycle disposition decisions.
3. Remove contents and equipment from buildings/facilities as necessary.
4. Demolish buildings/facilities.
5. Assess environmental media contamination and remediate as required.
6. Coordinate cleanup with community end use preferences including possible reuse of some facilities/buildings and recycle/reuse materials as practical.

Projected D&D milestones

- Award D&D contract, transition GDP part of site to D&D programme.
- Determine D&D and waste disposal method(s); put in place waste disposal programme.
- D&D GDP Process Buildings, Ancillary Buildings and related GDP site facilities.
- Remediate contaminated soils/subsoils, ground and surface water; define land access restrictions.

Current D&D/cleanup activities / completion date:

- X-633 Cooling Tower Complex / January 2011.
- X-533 Electrical Switchyard / October 2010.
- X-760 Chemical Engineering Building / October 2010.
- Surplus Uranium Materials Disposition / December 2010.
- X701B Groundwater Plume Source Remediation / September 2011.

Cost estimates

- Original Project Conceptual Cost Range = $5 B - $12 B, Completion Schedule = 2044.
- D&D Contract Profile Range = $250 - $325 M/year, Contract Period = 5 years, plus a 5 year option. (The initial D&D contract is expected to accelerate the project, but it is not anticipated to complete the overall project.)

Waste/recycle/reuse material

Projected decommissioning/demolition debris

D&D Waste:	1 700 000 m^3
Remedial Action Waste:	450 000 m^3
Total # Buildings/ft^2:	10 600 000 ft^2
Electrical Wiring:	4 600 miles
Process Piping:	600 miles

Unique attributes, contributions

The Portsmouth D&D is following the decommissioning of the Oak Ridge GDP. The Portsmouth project has an opportunity to incorporate lessons learned from the Oak Ridge project to improve performance.

A2.1.19 West Valley Demonstration Project (WVDP)

Note: The West Valley Project end state was originally Stage 2 and previously reported as Complete. This Project has become active once again with an end state of Stage 3.

Organisations:

- Nuclear Fuels Services, Inc. (NFS) (Reprocessing Facility Operator)
- New York State Energy Research and Development Authority (NYSERDA) (Reprocessing Facility Owner)
- U.S. Department of Energy (DOE) (Designated Remediation Lead Agency)

Description

Type of Facility: Commercial facility – reprocessed spent nuclear fuel.

Brief Description: Site area; 3338 acres. NFS began reprocessing spent nuclear fuel 1966 under license from US Atomic Energy Commission. Reprocessing duration 1966 - 1972. About 640 metric tonnes (705 tonnes) spent fuel reprocessed, resulting in 2.5 M liters high-level radioactive waste. Facility closed for modification 1972 – did not reopen, result; inadequately controlled legacy wastes. In 1980 US Congress directed DOE to conduct demonstration project to solidify waste; vitrification process completed in 2002; resulting in 275 storage canisters temporarily stored on-site.

WVDP was previously a USA/DOE project in the Cooperative Programme on Decommissioning (CPD) Programme in the late 1980's and early 1990s. Following the completion of limited initial facility decontamination and during the legal proceedings the WVDP was identified as a dormant D&D project within the TAG.

Operational details

Operational period

Spent nuclear fuel reprocessing: 1966 – 1972.

History

Western New York Nuclear Service Center (WNYNSC) started site 1961: located 30 miles south Buffalo, New York. NFS developed 200 acres of the site and operated the nuclear fuel reprocessing facility from 1966-1972, stopping reprocessing operations in 1972, with the intention of retrofitting the facilities to increase capacity. Resulting high-level radioactive liquid waste (HLW) volume was stored in two underground storage tanks. In 1976 NFS informed the State of New York (site owner) of its intention to terminate reprocessing activities and returned control to site owner upon expiration of its lease in 1980. Limited effective remediation, inadequate D&D and disposal controls resulted in burial of 78 000 m3 waste from commercial waste generators at a 15-acre State of New York licensed unlined landfill. Also NFS used a 7-acre US Nuclear Regulatory Commission (NRC) licensed unlined landfill. An uncontrolled air release (HEPA filter failure) and an acid leak released radioactive waste materials during reprocessing. These releases have resulted in cesium-137 contaminated soil (known as the “cesium prong”), and a strontium-90 groundwater contamination plume (known as the North Plateau Groundwater Plume). Additional contamination from radiological constituents is prevalent in the vicinity of the burial grounds, surface soils, buildings, water treatment plant and a number of holding lagoons.

In 1980 the West Valley Demonstration Project Act (Public Law 96-368) was enacted. Under the WVDP Act, the DOE was directed to take a lead role in solidification of the liquid HLW and decontaminate and decommission the West Valley facilities. DOE assumed control of 167-acres of site. In 1996 liquid HLW vitrification began and by completion of operations in 2001 a total of 275 stainless steel canisters containing vitrified waste had been processed. These canisters are presently stored on-site in a gutted cell in the former reprocessing plant. It was originally intended that the HLW canisters would remain in interim storage in this shielded storage cell pending transportation to a Federal high level radioactive waste repository. Recently, a Record of Decision (ROD) was issued to relocate the HLW canisters to a new on-site passive storage system. .

The 2006 NEA Co-operative Programme on Decommissioning described the completion of the vitrification activities and project progress in the areas of: 1) remote-handling waste facility construction, 2) cleanup of main plant head-end cells, 3) shipment and disposal of low-level waste, and 4) shipment of 125 spent nuclear fuel assemblies to Idaho National Engineering Laboratory for storage. In addition, DOE was preparing two Environmental Impact Statements (EIS); a Decontamination and Waste Management EIS for near-term activities and a Decommissioning and Long-Term Stewardship EIS.

In 2010 DOE and New York State Energy Research and Development Authority issued an Environmental Impact Statement (EIS) providing a comprehensive compilation of challenges and decisions made to remediate West Valley site. This EIS, titled “Decommissioning and/or Long-Term Stewardship at the West Valley Demonstration Project and Western New York Nuclear Service Center”, presented a comprehensive evaluation of four remedial action alternatives for the site. Remedial action alternatives included:

1. Sitewide Removal
2. Sitewide Close-in-Place
3. Phased Decisionmaking
4. No Action

Basic decommissioning strategy

Prior to implementing the decommissioning activities, WVDP plans to achieve an “interim end state” that is considered the starting point for decommissioning. This starting point was assumed for any of the EIS decommissioning actions.

With these conditions as the starting point, the alternative selected for decommissioning was Phased Decisionmaking; allowing for future actions to be based on phase results.

Phase 1 activities (estimated duration 10 years – with phase 2 decision within 10)

- Relocate HLW canisters to new storage area until Federal repository opened,
- Remove Main Plant Process Building and source area of groundwater contamination plume,
- Remove Vitrification Facility and Remote Handled Waste Facility
- Remove low level waste water treatment lagoons and systems
- Remove all ancillary facilities
- Remove soil contamination in the area around the Main Plant Process Building and low level waste lagoons,

- Ongoing management of the Waste Tank Farm; Tank/Vault Drying System, and NRC-Licensed Disposal Area.

Phase 2 activities (estimated duration 10+ years):

- Decommission HLW Tanks
- Decommission NRC-Licensed Disposal Area (if determined to be a DOE required action)
- Ship HLW to Federal Repository
- Decommission any remaining WVDP-related facilities

Decommissioning shutdown actions to date:

1. Buildings:
   - Industrial Facility Completions – 13 of 29 facilities.
   - Nuclear Facility Completions – 3 of 14 facilities.
   - Radioactive Facility Completions – 4 of 13 facilities.
2. Wastes:
   - HLW Packaged for Final Disposition – 275 of 275 canisters.
   - Liquid Waste Eliminated – 228 of 813.5 gallons.
   - Low-Level & Mixed Low-Level Waste Disposed – 27 986 of 29 899 m^3.
   - Transuranic Waste Eliminated – none disposed.

Interim end state

DOE and NYSERDA, in an ongoing goal of planning for site closure are taking actions to remove waste or facilities in order to achieve an Interim End State. The Interim End State will be the starting point for the EIS work. Activities to achieve the starting point have begun and will continue until completed. Major activities include:

- A number of minor, generally uncontaminated facilities will be closed, emptied of equipment, decontaminated as necessary, and demolished down to concrete foundations, floor slabs, or gravel pads.
- The Main Plant Process Building, except for the area used to store vitrified HLW canisters and the areas and systems that support HLW canister storage, will be decontaminated to a demolition-ready status. The 01-14 Building and the Vitrification Facility in Waste Management Area (WMA) 1 and the Remote–Handled Waste Facility in WMA 5 will be decontaminated to a demolition-ready status.
- A tank and vault drying system will be installed at the WMA 3 Waste Tank Farm to dry the remaining heels in the waste storage tanks.
- A permeable treatment wall will be installed in WMA 2 to mitigate further North Plateau Groundwater Plume migration. The North Plateau Groundwater Plume and background soils were sampled for potential hazardous constituents. These samples were also analyzed for radionuclide content.
- Waste created by activities to achieve the EIS starting point eventually will be shipped off site for disposal, with the possible exception of potential non-defense transuranic waste.
- An up-gradient barrier wall was installed, and a geo-membrane cover was placed over the NRC Licensed Disposal Area in 2008 to help mitigate surface water infiltration.

Cost estimates

$380 million (Amount budgeted for Fiscal Year 2009 through Fiscal Year 2013.)

Special hazards or challenges

1. Human Health Impacts.

2. Transportation Impacts.
3. Waste Management Impacts.
4. Long-term Human Health Impacts.

Waste

Estimated total volume of all waste classes = 1 600 000 m^3, however exact volume is dependant on Phased Decisionmaking alternative inputs.

Unique attributes/challenges

- WVDP presents a number of challenges and lessons learned, from political, technical and environmental perspectives. Clearly neither the West Valley site owner nor NFS, the spent nuclear fuels re-processor anticipated the complexity of the endeavor they were about to undertake. Likewise neither party had an awareness of the potential for the spread of radioactive materials into the environmental media at the West Valley site.
- Because Congress chose to conduct a DOE demonstration project on a state-owned site, DOE is operating on the site that has an NRC Part 50 license. DOE was only required to decommission certain facilities on the site, requiring the site owners to decommission all other areas under the Part 50 license. So while the NRC license remains in effect, the technical specification has been placed in abeyance while DOE is managing the site. NRC role at this time is one of consultation to DOE, not regulation.

A2.2 Category 2 fuel facilities and fuel related projects (dormant)

None at this time.

A2.3 Category 3 fuel facilities and fuel related projects (complete)

A2.3.1 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 decommis-sioning 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 1.48×1 010 Bq (4 Ci) including a number of difficult to measure nuclides like 63Ni. 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.

A2.3.2 AT-1, France

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

The decommissioning has taken place in five steps:

- Alpha cells.
- Small beta-gamma cells.

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

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

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

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

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

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

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

Clean up operations on the AT1 was completed by the summer of 2001 (stage 3 without civil works demolition). The facility had been radiologically declassified; entrance is permitted without any protection or control. The residual contamination on site is less than 37×10^{6} Bq (1mCi), which is the threshold for Installations Classified for Environmental Protection.

The facility, part of a basic nuclear facility in La Hague processing centre managed by AREVA, had been transformed as an example of what "it can and has been done".

A2.3.3 Active Chemical Laboratory (ACL), Sweden

The Active Chemical Laboratory at Studsvik, Sweden, and its associated ventilation and filter building were demolished in 2006 and are now in a Stage 3 status with only some of the waste waiting for a interim storage. The aim of the project was 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).

ACL were erected during 1959 to 1963 by the AB Atomenergi, the former research centre in Studsvik. The work in the laboratories came to an end on 31 December 1997. Six laboratories in area No. 1 were already decommissioned and accepted by the authorities as a free released area in 1990. The decommissioning work in these six rooms was performed during 1988 - 1990.

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

A pre-project had been conducted in 1998 to make a first radiological characterisation and, based on this, produce a decommissioning plan, with time schedule and estimated costs. The pre-project also included the remediation of the asbestos insulation on the site. The study indicated a cost of MSEK 16.8 over a three-year period. The period was reduced by the owners later to 2 years but became eventually 6 years. The re-planning of the project had been performed by a group of five people during two months. Demolishing was performed in 2006.

Half way in the project, the authorities issued conditions for the clean-up activities. The conditions stated that the project should: Perform reasonable clean-up activities, Show compliance with EU recommendations RP 113 (surface activity levels for demolishing) for the expected nuclides, Make nuclide specific measurements of residual activity, Search for spot activity if it is suspected that there are spots exceeding 150 Bq alpha or 1 500 Bq beta/gamma, Disregard naturally occurring activity that cannot be attributed to the operations performed in the facility, Take measures against re-contamination of cleaned areas.

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

This planning phase was executed with reduced personnel and on the basis of experiences acquired from area 1 as its basis. It resulted in a total cost for dismantling, decontamination and free release of the buildings of MSEK 70 (8M €). Waste measurements and collection are included in these figures but not processing, storage and disposal. Pre treatment of waste (melting and incineration) 7%, Cleaning 18%, Manual measurements 29%, ISOCS 17%, Consultants 7%, Management 16%, General purchases 6%.

Final waste cost estimates is extra (30 drums (200 L) of mixed waste, 41 tonnes of ash and 5 tonnes of ingots to interim storage).

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

Scrap metal was sent to the melting facility and consisted mainly of steel and aluminium. Approx 95% of the metals, 120 tonnes, were free released as ingots after the melting procedure.

Almost 50 tonnes of removed concrete, asbestos and insulation materials were deposited at a community dump. 20 000 tonnes of the building rubble was used to fill the hole in the ground.

Unique attributes/challenges

Each lab in the building had its own nuclide history so any common nuclide vectors could not be found.

- The decommissioning has been completed. All of the operations were conducted in the traditional manner using direct manual work by the operators.
- In one room in the cellar Am was found. Cracks in the concrete floor resulted in contamination in the soil under the floor.
- Only the mobile gamma spectrometry-technique was new for us. ISOCS was used for scanning of larger areas (~ 4 m^2). The aim was to detect gamma radiation and by the use of correlation factors make an estimate on alpha contents

- The choice of instrumentation has an effect on efficiency. Using hand-held instruments is time-consuming and demands a high degree of patience from the staff. It is also clear that detectors with a large detector area are more efficient than those with a smaller area. In-situ gamma spectrometry was not used from the beginning but is a good complement and should have been used from the start, as well as during pre-studies. Technical problems at the beginning, together with late deliveries, slowed down the project.
- The decommissioning ended up being a very educational and informative operation with techniques used in recent decommissioning.
- No significant incident occurred during the decommissioning.

New equipment

Mobile gamma spectrometry (ISOCS) were used for the first time in a decommissioning project. It is now compulsory in future decommissioning.

Licensing issues

Half way in the project, the authorities issued conditions for the clean-up activities.

Lessons learned

We could most certainly have gone faster, by waiting for the authorities issued conditions (set by 2001) and by not taking the dismantling project for an R&D testing ground.

A2.3.4 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 decommis-sioning 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)

A2.3.5 Fernald

Project identification

Project Name: Fernald Closure Project (FCP), also known as Feed Materials Production Center (FMPC).

Organisation: operated by U.S. Department of Energy (DOE).

Current Status: Decommissioning/Environmental Restoration completed 2006. Current use; nature preserve. Site and on-site waste disposal facility managed by DOE Office of Legacy Management.

Facility information

Type of Facility and Description

Produced high-purity, low-enriched uranium metals as production reactor feed materials for U.S. Department of Defense (DOD) Programmes. Site, 1 050 acres, operated 37 years; process operations wastes released into environmental media. Site underwent contamination characterisation and remediation. Current status; remediated with stewardship by DOE Legacy Management programme. Availability of funding to support accelerated schedule resulted in completion ahead of schedule and under budget.

Operational background

History

Processed uranium feed materials yielded high-purity uranium metals used at DOE or DOD facilities in U.S. weapons programmes from 1952 to 1989. FCP also processed plutonium and tritium feed materials for national defense uses and for use at DOE Hanford and Savannah River sites' production reactors. Manufacturing was in seven site buildings within 140-acre Production Area. During operations about 500 million (M) pounds (lbs.) of uranium metal products were produced. Site also was major U.S. depository for thorium-related nuclear products and recycled uranium used in reactors at Hanford. Manufacturing processes by-product solid and liquid wastes stored in impoundments on-site 1952 – 1984; after which all process wastes were containerized and disposed off-site. Production era contamination in soil, surface water and groundwater detected on-site and off-site. DOE initiated environmental investigations in 1986. Subsequently DOE initiated site remediation. U.S. Congress terminated site production mission in 1991.

Basic decommissioning strategy

Operational Shutdown Scope

At end of uranium processing era site mission changed to remediation. Equipment and process systems placed in a safe configuration; DOE, federal and state regulators, began remediation process. Thirty Removal Actions to stabilize site conditions and mitigate releases of contaminants into the environment completed. FCP site subdivided into 5 Operable Units (OUs): OUs 1 through 4 considered contaminant "sources" and OU 5 considered "environmental media" receptor from contaminant sources. DOE implemented comprehensive public relations programme to obtain input from all site stakeholders.

Decommissioning shutdown actions

At end of processing site contained inventory of: 1) 6.4 M ft3 containerized low-level waste, 2) 186 000 gal. liquid low-level mixed waste, 3) 31 M net lbs. nuclear product, 4) 255 process-related/administrative buildings, 5) three concrete silos containing 13 900 yd^3 of low-level radioactive mixed waste, 6) six waste pits containing more than 1 M tonnes (T) waste, and 400 acres containing 2.4 M yd^3 contaminated soil. Site generated an estimated 1.5 billion lbs. radioactive waste during its life.

The Remedial Investigation/Feasibility Study (RI/FS) process results, and subsequent Records of Decision (ROD), issued by federal courts, legally defined agreed-upon remedial actions for 5 OUs. Solid waste material, construction/demolition debris and excavated contaminated waste and soil, were dispositioned by a combination of on-site and off-site disposal. An On-Site Disposal Facility (OSDF) for emplacement of low-level contaminated material and D&D debris was constructed. Contaminated material exceeding On-Site Disposal Facility waste acceptance criteria (WAC) shipped off-site. Contaminated surface water remediated by erosion control and source removal. Groundwater contamination, a plume of uranium 223 acres in extent in underlying sole source aquifer was remediated by implementing long-term pump, treat, and reinjection system.

Time Scale

- Initiation of RI/FS Process – 1986.
- Shutdown of uranium materials processing – 1989
- OUs 1 thru 5 RODs (incl. OU 4 Amendments) – 1994 – 2003.
- On-Site Disposal Facility; First load-1997, Last load-2006, (Total Volume = 2 956 221 yd^3).
- Waste Pits Project, Unit Trains; First train-1999, Last train 2005(154 trains = ~737 000 T).
- OU 4 Waste Shipments by Truck to Disposal Sites – 2005 – 2006 (926 624 ft^3).
- Initiation of Contamination Plume Pump/Treat/Reinjection System – 1993.
- Anticipated aquifer restoration to 30 µg/l uranium clean-up goal – calendar year 2023.

Current Activities

- Site in DOE Legacy Management programme; Federal wildlife preserve with public access.
- Regular inspections of On-Site Disposal Facility (OSDF) cover.
- Sample and monitor OSDF leachate collection system and selected monitoring wells.
- Access restrictions at OSDF perimeter fence and monitoring wells.
- Ongoing Maintenance; well fields, treatment system, vegetation, erosion, access roads.
- Operate Contamination Plume Pump/Treat/Reinjection System for Aquifer Restoration.

Cost estimates

OU 1 Remediation (Waste Pits Area)

- 1994 ROD Adjusted Cost Estimate = $658M, Actual Cost = $449 M.

OU 2 Remediation [Non-Process Waste (Fly ash, Lime Sludge, Solid Waste) Disposal Areas]

- 1994 ROD Adjusted Cost Estimate = $42.3M, Actual Cost = $33.6 M.

OU 3 Remediation (Process Buildings, Facilities, Utilities & Structures)

- 1996 ROD Adjusted Cost Estimate. = $1,003.6B, Actual Cost = $522.7 M.

OU 4 Remediation (Silos 1 & 2 – Rad Waste, Silo 3 – Uranium Oxide Waste)

- Silos 1 & 2 1994 ROD Adjusted Estimated Cost = $55.6M, Actual Cost = $488.6 M.
- Silo 3 1994 ROD Adjusted Estimated Cost = $41.1M Actual Cost = $99.7M.

OU 5 Remediation (Construct/Operate/Close On-Site Disposal Facility, Remediate Soils & Waters)

- OSDF 1995 ROD Adjusted Estimated Cost = $539 M, Actual Cost = $224.2 M.
- Soil/Sediment 1995 ROD Adjusted Estimated Cost = $422 M, Actual Cost = $271.8 M.
- Restore Aquifer 1995 ROD Adjusted Est. Cost= $300.5M, Actual Cost (9/2006) = $218.6 M.
- Restore Aquifer *Projected Cost 10/2006 thru 9/2012 = $70.9 M.
- Restore Aquifer to Free Release Level – *Projected Cost 10/2012 - 12/2026 = $250.6 M.

  * Denotes Projected Aquifer Restoration Costs not included in project cost.

Total ROD estimated project cost =	$3,062.1 B
Total project cost =	$2,308.2 B
Total accelerated remediation savings =	$753.9 M
Total cumulative lifecycle cost (12/92-10/06) =	$4.4 B

Impacts of Schedule Acceleration:

FCP remediation demonstrates the significant impacts possible from successful project schedule acceleration. A well-managed project can achieve substantial reductions in schedule and cost. Proactive project controls, project management, and a motivated work force dedicated to safe and efficient operations are invaluable resources.

Project declared complete by contractor 10/2006; DOE accepts remediated site – 1/2007

Waste

Decommissioning/Demolition Debris

- OSDF Disposal [meets Waste Acceptance Criteria (WAC)] = 2,956,221 yd^3
- Off-site Disposal (> OSDF WAC) = 21,724 yd^3 (acid brick, lead flashing, Tc-99 contaminated concrete)

Uranium Process Wastes/Contaminated Soil/Subsoil Material

- Off-site Waste Pits Waste = 710 000 yd3, 1 053 300 T (moisture content reduced to meet WAC).
- Off-site Silos 1, 2 & 3 Waste = 926 624 ft^3.
- Off-site Soil/Subsoil Material = 186 392 yd^3.

Dismantling methods

- Implosions reduced time to D&D large, multi-story buildings and elevated water towers.
- Scaffolding used to remove large amounts of transite siding from buildings.
- Gas powered cutting torches used to more efficiently cut steel walled vessels and tanks.
- Use demolition equipment rather than manual labor where possible – examples; mechanical shears on buildings, internal piping, electrical conduits and contaminated material.

Other comments

Cost Savings Explanations

- Group structures into like D&D units; reduced number of engineering/construction packages.
- Greater emphasis placed on work planning; some work subcontracted using performance based, fixed-price subcontracting strategy; pre-certification of potential subcontractors.
- Greater emphasis placed on worker training – reduced potential for injuries, which reduced work stoppages and increased productivity; self-performed work where possible.
- Continuously observe excavations to identify materials, segregate, and avoid double handling.
- Plan excavation site dust and erosion controls; ensure particulate and contaminant controls.
- Larger articulated trucks more efficient than small articulated or road trucks.
- Central Waste Storage building was planned in the OU 3 ROD, however due to accelerated schedule, work and planning efficiencies building was not built.
- Acceptance by public and regulators of On-site Waste Disposal.

Unique attributes/challenges

- Releases of conventional and radioactive contamination into air, land, and water environmental media to both on-site and off-site receptors.
- Site's urban setting increased potential for contaminant releases to affect large population.
- Site's close proximity to nearby river and directly over drinking water sole-source aquifer.
- Use of asbestos-containing building materials commonly used at the time the facility was built.
- Logistics of safely handling and disposing of a waste stream ranging from conventional waste to low-level and highly radioactive waste, to usable nuclear materials inventory.
- Cleanup of contamination from inadequate air, land, and water pollution control, containment and treatment engineered design safeguards in place during production era.
- Limited development, knowledge, and history of available technologies to successfully clean up radioactively contaminated sites.
- Management of work force to transition from production to remediation.
- Providing work force with opportunities to enhance their futures after site closure.
- Encouraging stakeholder participation and assure them their inputs would be considered.
- Siting and installation of new well field and water supply to adjacent industries.
- Total rail shipments; 201 unit trains (12 000 railcars), no releases, transport incidents or injuries.
- Decontamination and removal of ~ seven miles rail tracks, crossings, switches, three yard locomotives, and 250 railcars (with lids) to other DOE sites for reuse.
- Stabilize radioactive waste and install radon control system in two deteriorating concrete silos.
- Remove, handle, slurry, containerize and transport radioactive wastes to disposal site.

New methodology developments

- Performing radioactive hazardous materials training courses to all governmental units (townships) emergency response staff along rail route in eight “transited states”.
- Negotiating shipping contract with rail carriers reluctant to transport radioactive wastes.
- Siting and construction of an OSDF over a designated Sole Source Aquifer.

New equipment or instrumentation developments

- Procured 250 lined lidded railcars; modified to meet container shipping requirements.

Licensing and political issues

- Negative publicity due to detection of radioactive contamination in water wells of adjacent industries and private residences.
- Interaction with regulatory agencies about extent and scope of characterisation and remediation and considering stakeholders’ input and schedule limitations due to uncertain funding availability.
- Consideration of effects to local, regional economy and site work force resulting from the change in the FCP’s mission from production to remediation, and ultimately to closure.
- Provided public water supply to residences with contaminated potable water wells.

Lessons learned

- Anticipate hazards due to changing site conditions; frequently update project risks.
- Emphasize quick practical solutions with worker buy-in; get worker buy-in on objectives.
- Emphasize constant adherence to radiation protection and contamination control processes; do not let accelerated schedule cause workers to take shortcuts.
- Use graded approach; match controls to hazards; “manage contract, not the contractor”.
- Measure and report safety issues at first aid and near-miss levels.
- Keep workforce engaged in understanding overall work scope hazards.
- Do not underestimate time needed for baseline and procedural preparation.
- Retain adequate personnel with site knowledge; implement “Cold & Dark” approach early.
- Understand how contract performance is being determined and reported. Use alternative performance indicators such as regulatory document close outs to track performance.
- Involve regulators essential for overall success; communicate at least once per week.
- Selective excavation has advantages: cost savings, and disadvantages: extended schedule.
- Utilize water mist/fogger/spray system to improve dust suppression during D&D.
- Accurately assess waste stream volumes as waste is generated.
- Use12-feet deep plus deep trenches around buildings to ensure isolation of utilities.
- Spread out critical time and sequence factors for work activities to prevent multiple critical paths from developing late in a project; do not underestimate value of site walk-throughs.
- Be direct, open and honest about objectives, challenges, constraints, progress and end states.
- Involve public early in process; be open and educate them to obtain early ‘buy-in’.
- Develop trust between Public/Regulators/DOE.

The project was completed 12 years earlier than anticipated and $7.8 B below initial estimates.

A3.1 Projects no longer participating in the programme

A3.1.1 Prototype Fast Reactor (PFR), United Kingdom

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

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

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

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

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

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

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

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

Other items of interest:

- The primary sodium is heated by electric heaters to keep it molten. They will no longer be needed when pumping starts.

- The 1 000 t of primary sodium contains only about 2 g of 137Cs. 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.

A3.1.2 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% 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.

A3.1.3 Paldiski, Estonia

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

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

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

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

The project is characterised by some unique features:

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

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

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

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