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Innovative and Adaptive Technologies in Decommissioning of Nuclear Facilities

Final report of a coordinated research project 2004-2008

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FOREWORD

There are dozens of old reactors and other nuclear facilities worldwide that are either being actively dismantled or are candidates for decommissioning in the near term. A significant proportion of these facilities are situated in Member States or institutions that do not have adequate expertise and technologies for planning and implementing state of the art decommissioning projects. The technology selection process is critical in that regard.

The main objective of the IAEA technical activities on decommissioning is to promote the exchange of lessons learned in order to improve the technologies, thereby contributing to successful planning and implementation of decommissioning. This should be achieved through a better understanding of the decision making process in technology comparison and selection and relevant issues affecting the entire decommissioning process. The specific objectives of the Coordinated Research Project (CRP) on Innovative and Adaptive Technologies in Decommissioning of Nuclear Facilities include the following general aspects: (a) To establish methodologies and data needs for developing concepts and approaches relevant to technology comparison and selection in decommissioning; (b) To improve and expand the database on applications and performance of various types of decommissioning technologies; (c) To address specific issues for individual decommissioning technologies and generate data relevant to their comparison and selection.

It is also expected that this project, and in particular the papers collected in this TECDOC, will draw Member States' attention to the *practicality* and *achievability* of timely planning and implementation of decommissioning, especially for many smaller projects.

Concluding reports that summarized the work undertaken under the aegis of the CRP were presented at the third and final research coordination meeting held in Rez, Czech Republic, 3-7 December 2007, and collected in this technical publication. Operating experience and lessons learned in full-scale applications, as well as key results in laboratory scale or pilot scale research and mathematical models, are among the most significant achievements of the CRP and have been highlighted.

The IAEA wishes to express its thanks to all the participants in the project and would like to take this opportunity to acknowledge the cooperation and warm hospitality of the institutions that hosted the RCMs. Special thanks are due to P. Dinner (IAEA) and A. Junger (IAEA) who reviewed the draft and prepared national papers for publication. The IAEA officer responsible for the CRP was M. Laraia of the Division of Nuclear Fuel Cycle and Waste Technology.

EDITORIAL NOTE

CONTENTS

1. INTRODUCTION

There are large numbers of old reactors, other nuclear facilities and legacy sites worldwide that are either being actively dismantled or are candidates for decommissioning in the near term. A good portion of these facilities are situated in Member States that do not have adequate expertise and technologies for planning and implementing state-of-the-art decommissioning projects. The technology selection process is critical in that regard. Currently, a global picture of decommissioning technologies shows that most decommissioning technologies are readily available in industrialized countries. This includes, but is not limited to: characterization, decontamination, segmenting, and related waste management. However it should be noted, first, that such technologies can hardly be deployed without consideration of and adaptation to the working environment (layout, radiation and contamination levels, temperature etc). Secondly, the selection process of alternative technologies available in the market is not a simple one (except for routine, standard applications) in that it involves consideration of a number of technical factors (performance, speed, waste generation etc) and managerial factors (direct and indirect costs, manpower, skills, hazards etc), and of the advantages and drawbacks of individual technologies. The final selection will generally be based on a case-by-case cost-benefit or multi-attribute analysis. A standardized approach in technology selection is currently applicable in a minority of cases only. Thirdly, there are a few decommissioning aspects where technologies still have to be further developed to achieve full maturity. To mention a few, this is the case of management of special materials (graphite, beryllium), very low level detection of radioactivity concentrations, and remote operation/robotics. The unique design of some older prototype facilities may add complications that can only be solved on a case-by-case basis at the decommissioning stage. As a general point for industrialized countries, those responsible for the decommissioning of nuclear facilities may be reluctant to promote innovation if they do not see commercial advantage.

The picture from Member States having limited resources or scarce expertise in decommissioning is quite different. In such circumstances, decommissioning operators have to struggle with constraints additional to those faced in developed countries and may have to opt for less than “state of the art” solutions. In many cases, there is clearly a desire for individual Member States to develop their own decommissioning technologies for use in their organizational and regulatory arenas. In part, this is due to the need to understand the effects of decommissioning under site specific conditions in order to satisfy the nuclear regulators, but also it is due to the fact that many available processes are proprietary formulations and expensive to buy in the open market. In some Member States, it is very difficult to implement full decommissioning for these reasons and the costs associated with such a project are relatively high. Achieving the proper balance between development of project and country specific technologies (purportedly at the lowest cost) or purchasing technologies in the open market, remains a serious challenge in many countries. Timely allocation of decommissioning funds is important to alleviate these concerns and minimize delays in project implementation.

The IAEA has for years provided practical and regulatory guidance on decommissioning with the objective of fostering exchange of information and know-how and harmonizing approaches and strategies. To this end, mechanisms such as dissemination of documents and reports [1–8], training courses, or direct assistance to Member States [9, 10] have been applied. Another useful mechanism is the coordinated research project (CRP), which is relevant to the work presented in this document, and will be described in the following sections.

2. COORDINATED RESEARCH PROJECTS ON DECOMMISSIONING

Although the state-of-the-art technology for decommissioning nuclear reactors is probably adequate to cope with most difficulties associated with the dismantling of such facilities, it is generally necessary to improve, adapt or optimise technologies for the specific needs of the reactor to be dismantled. Also, it may be possible in many cases to develop or adapt simpler decommissioning technologies rather than purchase costly equipment, e.g. remote handling equipment. Learning from others rather than re-inventing the wheel makes sense in today's global context. This approach would probably match the needs of many developing Member States. In general, research and development of decommissioning technologies is an active research field. Exchanging conceptual information and know-how is the very *raison d'être* of a CRP.

This CRP on Innovative and Adaptive Technologies in Decommissioning of Nuclear Facilities represents the continuation of two CRPs conducted earlier, in 1989–1993 and 1997-2001 in the field of decontamination and decommissioning of nuclear facilities. The main results of these CRPs were collected in TECDOCs for distribution to Member States [11, 12]. As decommissioning covers a broad, multi-disciplinary field, it is widely accepted that to be cost effective, a CRP should be focussed on specific technical topics, such as decision-making in selection of decommissioning technologies, as in this case, and/or specific types of nuclear installations (such as research reactors [11]).

3. SCIENTIFIC SCOPE AND PROJECT GOALS

The overall objective of this CRP is to promote R&D activities, as well as the exchange of information and transfer of knowledge, in order to improve the technologies that are important in the planning and implementation of decommissioning. This may be achieved through a better understanding of the decision-making process in technology comparison and selection, as well as in addressing specific and relevant issues affecting the entire decommissioning process. The objectives of this CRP include the following general aspects:

- To establish methodologies and data needs for developing concepts and approaches relevant to technology comparison and selection in decommissioning
- To improve and expand the database on applications and performance of various types of decommissioning technologies
- To address specific issues for individual decommissioning technologies and generate data relevant to their comparison and selection

The following aspects were considered in detail:

- Planning of decommissioning activities with a focus on interactions with relevant technologies;
- Identifying technological needs, constraints and priorities;
- Exploring market availability of technologies;
- Gathering experience on technologies from other decommissioning projects;
- Evaluating costs and financing of technology procurement or developments;
- Identifying proprietary aspects of technology and methodology and their impacts on the decision-making;
- Conducting cost-benefit or multi-attribute analyses of specific cases of technology comparison and selection;
- Searching the market for adaptive technologies;

- Conducting R&D on innovative technologies;
- Identifying training requirements and performing training for decommissioning technologies; and
- Capturing and elaborating on operating experience and lessons learned.

The process of selecting and awarding agreement/contracts was completed by mid 2004. Three contracts (Brazil, Cuba, and Russian Federation) were awarded, as well as ten research agreements. After the first Research Coordination Meeting (RCM) (Halden, Norway, 4–8 April 2005) a second research agreement was granted to Republic of Korea on a new topic. The second RCM was held at Keswick, near Windscale, United Kingdom, 13–17 November 2006, and the third and last RCM at Rez, near Prague, Czech Republic, 3–7 December 2007. According to a rough categorization, the CRP involves fully industrialized Member States, Member States with limited resources and little experience in decommissioning and Member States presenting a mixed picture as far as decommissioning resources and experience is concerned. All participating Member States, however, had a strong interest in expanding their capabilities and resources in nuclear decommissioning. The following Member States finally took part in the CRP: Argentina; Austria; Belgium; Brazil; Cuba; Czech Republic; Denmark; Republic of Korea; Norway; Russian Federation; Slovakia; Ukraine; and United Kingdom.

4. SUMMARY OF MAJOR TECHNICAL ACHIEVEMENTS

As noted above, this CRP investigated a very broad range of decommissioning-related activities and technologies. National approaches presented in the CRP were appreciated by the participants as being generally of high quality. The following highlights specific R&D areas covered by the CRP and some of the pertinent individual projects.

Decision-making tools employed can be grouped into two main categories:

- Those represented by R&D on decision-making approaches for selection and optimizing decommissioning technologies and procedures for entire projects (e.g. Republic of Korea, Slovakia) and

- Those represented by R&D (application) of “generic” tools and problem-solving techniques (e.g. Norway, Slovakia, Belgium, Republic of Korea).

Aspects relevant to and facilitating (or hindering) decommissioning were investigated in the national context of the Ukrainian project, which is still at the planning stage. Planning aspects were also described in a number of other decommissioning projects.

Characterization - both physical and radiological - was extensively performed in and reported by the UK project. Due to the unknown structural damage caused by the 1957 accident, intrusion into the Windscale Pile 1 structure was initially forbidden by the UK regulators. Therefore, during the initial stages of the CRP, most information relevant to characterization and the planning of decommissioning had to be gained using indirect means. Thus decision-making in such uncertainty is the key element of the UKAEA contribution to the CRP. A new Windscale Pile 1 safety case was completed in June 2006. It came to the conclusion that the risk of criticality, thermal effects, and other hazards could be excluded in case of a reference seismic event. On that basis, UKAEA has submitted a safety case to support the request of intrusive characterization into the damaged reactor core. This safety case has now been approved and more detailed characterization of the fire-affected zone of the reactor has been made possible.

A wide range of decontamination technologies were investigated in Argentina and Russia, and were organized and "catalogued" by the latter. Among other projects, the Brazilian contribution focused on detailed tooling-development for management of radioactive wastes resulting from decommissioning of nuclear fuel cycle facilities.

Dismantling technologies were addressed in a number of projects. In Austria, in the course of the CRP, the Astra reactor was completely dismantled, successfully illustrating a precise correspondence between characterization and waste-generation ending with the reactor building's release for unrestricted use. Republic of Korea's KRR-2 is fully dismantled now. This project also focused on lessons learned from the application of project management techniques. In Belgium, during the CRP, the BR-3 project moved significantly towards completion. Decommissioning of DR-1 in Denmark was completed in the course of the CRP and the site released for non-nuclear use. Decommissioning of DR-2 was also planned and completed in the course of the CRP and included assessment of alternative techniques.

The Czech project addressed ongoing decommissioning activities at Rez Nuclear Research Institute. It focused on construction, which was hardly conducive to smooth decommissioning, and the need to develop unique solutions.

A number of R&D decontamination and decommissioning activities were extensively described on a national scale in a project from the Russian Federation, to create a database of waste management and decommissioning experience, technologies and infrastructures available in that country and applicable to future decommissioning projects.

It is noteworthy that several projects highlighted decision-making conducted with a scarcity of resources. For example, the Cuban project ended in a condition of restricted release of the site, since obtaining unrestricted release would have been beyond the economic means available.

Norway's Institute for Energy Technology (IFE) has been hosting the OECD NEA Halden Reactor Project (HRP) since 1958, and performs research on fuel and safety issues both within HRP and on a bilateral basis. Halden Virtual Reality (VR) Centre focuses on the Man Technology Organisation (MTO) domain both in the nuclear and other areas, building on IFE's vast experience in applying advanced graphical visualization technologies and human factors to real-life challenges. Their VRdose system is interactive software intended for simulation and optimization of outage, maintenance and decommissioning activities, with special focus on raising radiation-exposure awareness. R&D on VR and its deployment in decommissioning represented IFE's contribution to the CRP.

The Slovak code OMEGA is capable of performing decommissioning assessments– including selection of alternative technologies – iteratively to reach the desired level of accuracy,. For example, OMEGA is available to make cost estimates from a very preliminary to a detailed level. In fact the detail of cost estimates depends to a large extent on the detail of decommissioning planning, both being of an iterative nature.

This CRP also addressed a variety of nuclear facilities. Research reactors include Astra (Austria), BR-3 (Belgium), DR-1 and DR-2 (Denmark), and KRR-1 and KRR-2 (Republic of Korea). Activities at nuclear power plants or production facilities were addressed for NPPs in Argentina and Windscale Pile 1 (UK). The decommissioning of nuclear fuel cycle facilities was addressed by the Brazilian project. Small medical, industrial or research facilities in Cuba were addressed by another project. The remaining projects addressed a variety of national facilities or tools of generic applicability.

Some of the decommissioning projects developed under the umbrella of this CRP were active over the CRP time frame and their practical results were reported at RCMs (Astra, Austria; BR-3, Belgium; Brazilian, Cuban, Czech, and Russian facilities; DR-1 and DR-2 (Denmark); KRR-1 and KRR-2 (Republic of Korea); WWR-type research reactors (Czech Republic, Ukraine) and Pile 1, (UK). Other decommissioning projects were at the planning stage (Atucha 1, Argentina, Ukrainian facilities). As said, a few projects were intended to develop decision-making tools of generic applicability, with applications to specific installations exhibited (for the Norwegian project, applications in Italy, Japan, Russian Federation and Ukraine; for the Slovak project, applications to A-1 NPP, Slovakia and Studsvik facilities in Sweden).

5. STATE-OF-THE-ART AND PENDING ISSUES

Although the decommissioning industry cannot yet be regarded as fully mature in terms of delivering a standard package, the key elements of strategy development, waste treatment, dismantling and release from regulatory control have been separately demonstrated as achievable. As a result, with the implementation of the right organization and improved technology, the risks are being reduced. As more decommissioning projects are delivered, the risks will be reduced further. However, for some nuclear facilities it is still necessary to have solutions related to special problems such as the management of graphite and sodium materials or characterization of alpha and weak-beta- contaminated waste.

The main activities associated with decommissioning do not necessarily need to be as sophisticated as the technologies used for the construction of the plant. They need only be adequate to achieve the desired objective of decommissioning the facility or site. It is important to use proven methods which will provide secure planning and costing, rather than theoretical approaches relying on advanced technology to deliver what is only a potential improvement. Lessons learned through current or completed projects are available in the literature e.g. [13], but nothing can replace the user-to-user sharing of experience: the very epitome of a CRP.

Suitable dismantling and decontamination techniques exist for virtually all aspects of decommissioning. From the large number of decommissioning projects in progress or already completed, the conclusion can be drawn that conventional, robust methods and commercially available technologies can be used almost everywhere. Especially for smaller facilities with a low activity inventory and correspondingly low dose rates, the adaptation of techniques from conventional industrial applications provides good solutions in most cases. Where possible, tools and equipment available from the facility's operation might usefully serve new decommissioning applications. As one example, the ASTRA reactor at Seibersdorf, Austria, was recently dismantled and the concrete bioshield was diamond cut into segments weighing just under 10 t because of the capacity limits of the overhead crane [4].

Although decommissioning is a mature industry, innovative or substantially modified techniques will sometimes be needed in the future. One such example is the decommissioning of the Russian TVR reactor at the Institute of Theoretical and Experimental Physics, where one of the most radiation-intensive operations was the removal of horizontal channels. As the dose rate in the area of these channels was very high, a special remotely-controlled machine equipped with a crown milling cutter was developed, manufactured and successfully used in dismantling operations [4].

5.1. Segmenting/cutting techniques

Experience indicates that for the immediate future there is no general purpose segmenting/cutting method that can be recommended for all segmenting tasks. Thermal techniques, though generally small and easy to apply, usually require substantial effort for air filtering and contamination control for the aerosols produced, which makes them unsuitable for a number of segmenting tasks in contaminated areas. On the other hand, mechanical tools such as reciprocating saws generate reaction forces and produce larger sized particulates, which are very easy to control but cannot be used in confined spaces. Slightly different considerations apply if segmenting can be done underwater. Many thermal and mechanical cutting techniques have been developed for application in underwater nuclear decommissioning. In this case, thermal cutting techniques require substantial water cleaning systems.

As a significant example of progress, diamond wire cutting can now be efficiently applied to a large range of materials, including composite materials (Belgium). Segmenting or cutting techniques for decommissioning have advanced to such a state that only minor development of certain techniques is usually required to suit the individual needs of decommissioning projects. However, a number of tools may still need case-by-case adaptations, as outlined above. One important example is the need to adapt available equipment options to the specific requirements for preventing emissions of radioactive aerosols when working on contaminated materials.

5.2. Decontamination techniques

There is no universal decontamination technique because the performance of any given solution will depend on a number of plant specific parameters and requirements. Although decontamination techniques are generally available off the shelf, special consideration has to be given when planning their use in specific cases. Depending on the size of the facility in question, the costs and time required for installing and operating suitable decontamination techniques vary. Aspects to consider include the costs for management of secondary waste from decontamination and costs that can be saved by the resulting “downgrading” of the material suitable for decontamination from a higher to a lower category of radioactive waste (or even to conventional waste after clearance). Such an analysis also takes into account the fact that some material will require significant preparation before decontamination can be applied (e.g. pipes may need to be segmented to make inner surfaces accessible). Such an effort will only be worthwhile if the amount of material that can be salvaged in this way is sufficiently large. The break-even point will depend, of course, on country or plant specific conditions.

Decontamination techniques still leave some room for improvement. As of today, the physical-chemical process of decontamination is not fully understood and most of the decontamination processes are still partly based on trial and error. Chemical techniques, which are easier to use than mechanical techniques in small applications, still produce a significant amount of secondary waste. Reduction of this secondary waste can be achieved by improved regeneration of the decontamination chemicals. There is also scope for the waste treatment plants to be adapted to take these secondary wastes. In addition, experience indicates that more work may be required on radioactively contaminated concrete, addressing characterization methods, low dose decontamination methods and improved volume reduction for secondary wastes resulting from concrete decontamination.

As mentioned above, R&D in decontamination remains active. For example, molten salt decontamination (Brazil) was introduced in this CRP as one of the few first trials worldwide using this technology. The Argentinian project was intended to develop traditional vibratory vessel technology making use of national industrial capabilities.

5.3. Radiological characterization techniques

On the whole, sampling equipment is now well developed and is often based on equipment used in the non-nuclear field, such as diamond tipped core-drills used for sampling concrete and graphite (Astra reactor, Austria). Some additional developments have been undertaken on material containment systems and on techniques for minimizing secondary waste production. While established techniques for sampling contaminated and activated surfaces and materials are available, new techniques are emerging for specific applications.

Examples of some recent characterization techniques which have the potential for further development can be summarized as follows:

(a) Systems for superimposing radiation readings and spectrographic information onto visual images of an object.

(b) Methods for simulation of decommissioning activities by plotting these against positional data. Positional data can be provided for indoor situations by modified surveys, or outdoors by means of global positioning systems. Data can be displayed in the form of a CAD image of the survey area or a geographical map. CRP-related activities at Halden, Norway (VR) and Mol, Belgium (VISIPLAN) belong to this category.

(c) Methods for inserting radiation probes into pipes.

(d) Methods for automated collection of a large number of surface contamination readings.

(e) Extensive use of in-situ gamma spectrometers.

(f) Increased experience and instrumental sensitivity in detecting very low contamination levels, e.g. close to clearance levels.

(g) Broader identification of radionuclides, including those that are difficult to measure, by the use of radiochemical separation and fingerprinting techniques.

In radiological characterization, a market has evolved over the past decade. This not only includes manufacturing companies that are highly innovative and offer a wide variety of measurement devices, but also extends to contractors offering a total release measurement service. This may be useful in the future for those decommissioning projects that have a small waste inventory and where the specific purchase of state-of-the-art characterization devices may not be economically feasible.

5.4. Restricted vs. unrestricted release

In line with the anticipated “nuclear renaissance” and the increasingly competitive energy sector, optimization of decommissioning cost is becoming imperative. One of the aspects that came to light in this CRP is the opportunity of releasing facilities for restricted rather than unrestricted use. This is particularly relevant when available resources are small and all possible means must be exerted to make such resources sufficient. The Cuban project highlighted that restricted release can provide a satisfactory project end-state while ensuring fulfillment of radiological criteria for workers and the public.

5.5. Tools to support planning and decision-making

As noted above, the CRP dealt not only with case-by-case determination of the optimal decommissioning approach or technique, but also with generic decision-making methodologies and tools. Within the CRP, the Norwegian project stressed the use of interactive, animated software simulating decommissioning activities with a view to selecting the optimal strategy. One similar tool is VISIPLAN developed by SCK/CEN, Belgium, which was widely used in the context of BR-3 decommissioning. The Omega code, developed in Slovakia, is a powerful software allowing a multitude of parameters to be changed in a decommissioning plan in order to compare results (e.g. costs, waste generation) and evaluate the robustness of the strategy to variable inputs. This parametric approach is invaluable in taking strategic decisions.

6. PROJECT OUTCOME

The participants in this IAEA project supported the view that it had succeeded in transferring information and know-how from active decommissioning projects to those planning for decommissioning. It is also expected that this project, and in particular the papers collected in this TECDOC, will draw Member States’ attention to the *practicality* of timely planning and implementation of decommissioning. In some Member States there are nuclear facilities which are kept in an extended state of shutdown, pending decisions on continued operation, extensive refurbishment or decommissioning. This situation — which frequently lasts for many years —weighs heavily on staff morale and motivation, state resources and entails deterioration of structures and components, which may in the longer term have very serious safety implications.

The results of this IAEA project will offer many Member States the opportunity to move forward in their evaluation of the financial and other impacts of decommissioning their nuclear facilities, so that decommissioning actions can be initiated without undue delay. Aspects such as fuel and waste management and provisions for other technical, administrative and financial resources require timely preparation, and they all involve the knowledgeable selection of appropriate technologies.

In more general terms, the project will contribute to enhancing Member States’ overall project-organizational capabilities. As decommissioning is a multi-disciplinary process, the project will stimulate Member States to develop an integrated approach to decommissioning by making optimal use of resources available both domestically and internationally. In this regard, the project impact may go far beyond the scope of decommissioning techniques.

7. CONCLUSIONS

Given the fact that the need for decommissioning and environmental restoration exists on all continents, cleanup and restoration operations will tend increasingly to be of an international nature. There are three modes of international co-operation that can be utilised in this domain. The first is through bilateral arrangements between countries and/or organizations. The second is co-operation on a regional level and the third is through the activities of international organizations. The latter form of co-operation, with emphasis on information and technology exchange, including joint research and development and demonstration projects, has been very successful in the decommissioning area. Coordinated Research Projects are the typical mechanisms for implementing such a strategy. Cooperation of this nature has many benefits and is practical for several reasons. First, it makes good economic sense to share and learn from each other's experiences and compare future strategies. The resulting benefit is that it prevents duplication of efforts. A second point worth mentioning is that projects initiated by any or all of the international organizations tend to be considered more credible and therefore generate more financial support. Third, joint projects create a support network and a system of formal and informal peer reviews. This external review process enhances and adds technical credibility and validity to national approaches and methodologies. And finally, co-operation and exchange of information are required and used by countries as a means of checking their own progress — a means of calibration. As detailed in the accompanying national papers, a CRP is also a means for participating institutions to establish bilateral or multilateral contacts bound to bear fruit independently of and extending beyond the CRP framework.

When a CRP such as this is concluded, it is important to maintain momentum gained in the sharing and transfer of practical decommissioning information and retain the "network" links formed in the CRP. The newly initiated IAEA International Decommissioning Network (IDN) provides a vehicle to sustain the benefits described in the preceding paragraph.

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ANNEX
EXAMPLES OF NATIONAL EXPERIENCES

The examples provided in this annex cover a variety of topics, from simple, standardized technical practice to complex computer programs for optimizing the overall decommissioning process. It is believed that all these aspects are useful for providing practical guidance and information on how decommissioning projects are planned and executed in various Member States with a view to illustrating how technology and practice can be adopted from one decommissioning project to another. The examples given are not necessarily best practices, nor has their consistency with the IAEA's guidance been tested in detail. Rather they reflect a wide variety of national policies, social and economic conditions, nuclear programmes and traditions. Although the information presented is not considered to be exhaustive, the reader is encouraged to evaluate the applicability of these cases to a specific decommissioning project. Data and statements provided by national contributors are not necessarily endorsed by the IAEA.

ARGENTINA

DEVELOPMENT OF DECONTAMINATION TECHNOLOGY FOR TUBULAR COMPONENTS

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Abstract
The objective of this project is to explore the feasibility of applying conventional vibratory vessel technology for decontamination of radioactively-contaminated and/or chemically-contaminated materials such as pipes, metal structures, and others components. Tests with radioactively-contaminated materials of three aluminium tubes and one stainless steel bar were performed in a decontamination lab after preliminary cold tests of different materials. Tests showed that it is possible to clean both the external and the internal surface of contaminated tubes. Results show a decontamination factor around 10 after the first 30 minutes of the cleaning time

1– Introduction

Abrasion processes in vibratory tumblers are widely used in the manufacture of metals, ceramics, and plastics to smooth, clean, and polish the materials. The system is based on a mechanical action. Samples to be treated, solid abrasive media and liquid media are set up into a metallic vessel. This vessel is mounted on springs and lined with a suitable polymeric material. A special heavy duty vibratory motor provided with adjustable counterweights is attached to the vessel. Vibration is transmitted from the motor to the vessel putting the entire load in motion at the same time so that the abrasive media acts against the parts throughout the complete mass. Liquid media plays a major role in the process, helping to keep parts and media clean, inhibit corrosion and suspend tiny abrasive particles, among other functions. In many cases liquid media also contains surfactants, detergents and/or etching agents. Before working with contaminated material, cold tests were performed with different materials. Contamination was simulated by controlled oxidation at high temperatures. Samples of carbon steel, stainless steel, titanium, Zircaloy-4 and aluminium were used. Hot tests were perfomed with three aluminum tubes and one sample of stainless steel bar. Results are presented in this technical document.

2 – Equipment

For this project two vibratory tumblers were installed in the Constituyentes Atomic Centre: a toroidal laboratory machine and a rectangular industrial machine. The design and construction of the machines were provided by Vibro, an Argentinian private company.

2.1 – Laboratory machine

This is a small toroidal machine (Fig. 1), with a load capacity of 100 kg of abrasive media. Technical features of the machine can be summarized as follows:

Supplier	Vibro S.A.
Dimensions	Ø 850 mm x 1100 mm
Vessel Volume	120 Litters
Power [Hp]	1

Fig. 1: Laboratory Machine.

2.2 – Industrial machine

Specially designed to process large samples, this machine (Fig. 2a and 2b) can be loaded with 600 kg of abrasive media. The features are the following:

Supplier	Vibro S.A.
Dimensions	1600 x 600 x 700 mm
Power [Hp]	8

Fig. 2a: Industrial Machine.

Fig. 2b: Counterweight Setting.

3 – Abrasive media

Erosion of the treated material results from the repeated impact of the abrasive particles on the surface. The abrasive effect is not the only role the solid media plays in the process. Another primary function is to keep the parts separated during processing, in order to avoid them to crash against each other. The volume ratio of media to parts determines the degree of parts separation; at high ratios, parts are well separated and have little contact. This is an important consideration when part finish is critical. Although each case requires a proper process design to get the best results, the following table is provided as a reference guide regarding load composition.

Material	Samples (In volume)	Media (In volume)
Iron-based alloying	50 %	50 %
Non-ferrous alloying	30 %	70 %
Plastics	50 %	50 %
Ceramic and Glass	25 %	75 %
Wood	40 %	60 %

The main abrasive media parameters to be taken into account are shape, size, weight and abrasive properties of the material. The intensity of the eroding effect is related to the above described properties of the abrasive media, as well as the magnitude of the imposed vibration. Time of treatment is also to be considered: the longer the treatment, the greater the amount of material removed.

Solid media can be divided into two groups, depending on the role the media plays in the process. The first group is the abrasive media itself. Abrasive materials such as aluminum oxide, silica, SiC, etc. are bound in a polymeric or ceramic matrix forming the pieces used in the cleaning process, which are also known as “Chips”. Chips can have a variety of shapes including tri-star, bead, pill, or crushed particles (as shown in Fig 3a). The importance of the shape and size of the abrasive media rests on the possibility of the media of reaching the most intricate parts of the treated sample.

Fig. 3a: Commercially available Chips.

It is worth noticing that the specific gravity of ceramic media is almost twice as much of that of plastic media and the hardness of ceramic is much greater than that of plastic. Therefore, plastic media will be chosen for more gentle treatments where deformation of the parts treated must be avoided, whereas ceramic media should be selected if the eroding effect is to be maximized. Plastic deformation of the treated parts must be considered not only for the damage to the samples but also for the difficulty of cleaning the bottom of the deformed area. Once a pit is created by the impact of a too-energetic chip, the internal radius of the pit is usually smaller than the chips that are being used, therefore this area cannot be properly cleaned. This effect is called “Hammering effect” and must be avoided if the sample is to be fully cleaned.

The auxiliary media is the second group of solid media. Polishing and drying materials are important process aids. Stainless steel balls, ball cones or satellites, and stainless steel pins are used for burnishing and de-burring of ferrous and non-ferrous parts. Corn cob drying media is used in vibratory and rotary dryers. Glass beads are also used for polishing. In addition, adhesion prevention granules can be used to prevent flat pieces from sticking together. Commercially available samples of these materials are shown in Fig 3b.

Fig. 3b: Auxiliary Media.

4 – Tested materials

4.1 – Cold tests

Before working with contaminated material, cold tests were performed with different materials Contamination was simulated by controlled oxidation at high temperatures. Samples of carbon steel, stainless steel, titanium, Zircaloy-4 and aluminium were used. Oxide layers were evaluated before and after treatment by means of digital camera pictures, optical microscopy and SEM examination. Aluminum, zircaloy and titanium plates were electrochemically oxidized. As a result of this process, a thin and uniform surface layer was obtained. In order to follow the evolution of the process, observations were made at regular periods of time (Fig. 4a and Fig. 4b).

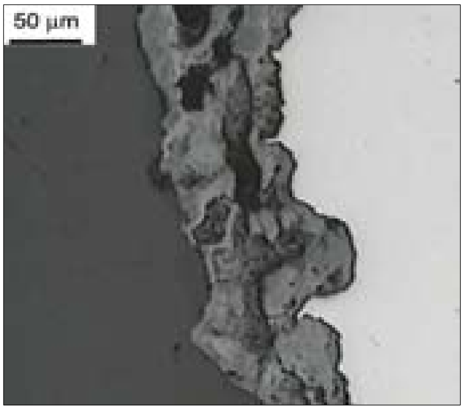

Fig 4a: Sample before treatment.

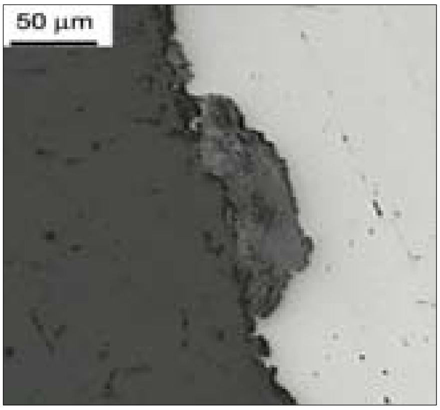

*Fig 4b: Sample after 30 minutes treatm*ent.

Also, a 500 mm long piece of the upper part of a non irradiated coolant channel tube of the Atucha 1 NPP was oxidized for two hours at 600 ºC. The oxide layer obtained was clearly identified due to its characteristic interference color (Fig. 5). The estimated thickness for this layer was 0.1 mm.

Fig 5: Upper part of Atucha 1 NPP coolant channel after oxidation process.

Once oxidized, the tube was treated in the industrial machine using cone-shaped plastic chips with a mean size ranging from 10 to 50 mm. Abrasive content of the chips used was 70 % aluminum oxide. Liquid media was a commercially available formulation especially designed for stainless steel provided by Vibro S.A with a pH value of 5.

Initial cleaning started after 10 minutes of treatment. After one hour the sample was rinsed and evaluated. External and internal surfaces of the tube were successfully cleaned.

(A) (B) (C)

Fig. 6: Upper part of an Atucha 1 Coolant Channel. (A) As received. (B) After high temperature oxide growing treatment. (C) After cleaning process.

Another important material tested was a carbon steel bar of 1inch diameter. The bar was oxidized for 15 days. As a result, a thick oxide layer appeared as shown in Fig 7.

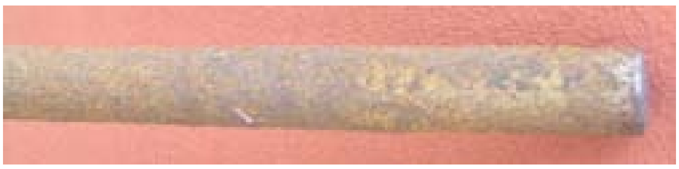

Fig. 7: External appearance of carbon steel bars after oxidizing process.

The bar was treated in the industrial machine and evaluated at the same treatment times used for other materials (6, 15, 30 and 60 minutes).

A **B** **C** **D**

Fig 8: Evolution of the cleaning process for A: 6 minutes, B:15 minutes, C: 30 minutes and D:60 minutes.

Also, a 100 mm OD, 3 mm wall thickness aluminum tube was used to test the ability for cleaning internal parts of the pipes. Three windows, 60 mm long, were cut along the perimeter and covered with a slightly larger piece, that was attached by bolts to each window, as shown in Fig 9.

Fig 9: Frontal and internal view of the device used to test ability for cleaning internal areas.

This device allowed to test three different materials at the same time, by replacing the detachable plates. A 2 mm thick aluminum plate, a 1.7 mm zircaloy plate and a 0.5 mm thick titanium plate were tested in the industrial machine using this device. The materials (Aluminum, zircaloy and titanium) were electrochemically oxidized. As a result of this process, a thin and uniform surface layer was obtained. The good adherence of the resulting oxides was desirable for testing the cleaning process. Fig.10 shows the oxidized samples before treatment and its characteristic interference colors.

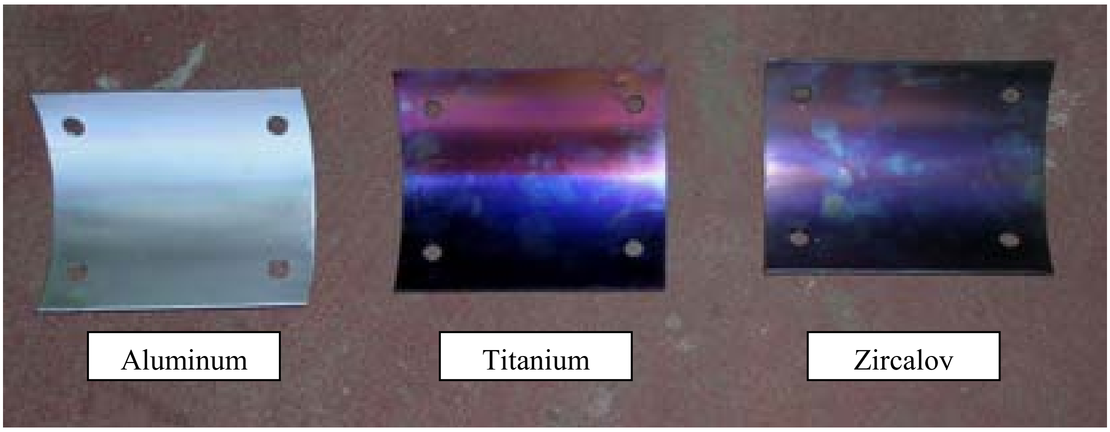

Fig 10: Electrochemically oxidized plates before cleaning process.

After surface treatment, samples were tested in the industrial machine. In order to follow the evolution of the process, observations were made at regular periods of time.

Results can be summarized as follows:

Material	Time for cleaning internal surfaces	Time for cleaning external surfaces	Suggested treatment time
Aluminum	60	6	60
Titanium	30	30	30
Zircaloy	30	6	30

4.2 – Contaminated materials tests

Two different kind of contaminated materials were tested in this work, three aluminum tubes (Figs. 11a and 11b) and one sample of stainless steel bar (Fig. 12).

Fig. 11a Aluminum tube outside view.

Fig. 11b Aluminum tube inside view.

Fig. 12: Stainless Steel bar.

5 - Characterization

Before and after 6 hours of tests radionuclide qualitative determinations were performed on samples 1 to 4. Gamma spectrometry of high resolution was the analysis method.
The following table shows the radionuclides detected during the first and second determination (before and after) and the counts rate of each determination.

Sample	Radionuclides Detected		Counts Rate	
	First Determination	Second Determination	First Determination	Second Determination
1	**^{60}Co - ^{137}Cs**	**^{60}Co - ^{137}Cs**	**0,48 - 59,28**	**0,05 - 3,00**
2	**^{22}Na - ^{60}Co - ^{137}Cs**	**^{60}Co - ^{137}Cs**	**0,01 - 1,24 - 133,36**	**0,07 - 5,35**
3	**^{60}Co - ^{137}Cs**	**^{60}Co - ^{137}Cs**	**0,82 - 55,53**	**0,01 - 4,00**
4	**^{60}Co - ^{134}Cs - ^{137}Cs**	**^{60}Co - ^{134}Cs - ^{137}Cs**	**2,59 - 0,22 - 4,84**	**0,09 - 0,002 - 0,04**

6 – Results

6.1 – Decontamination

Figure 13 shows the sample position measurements during the decontamination process.

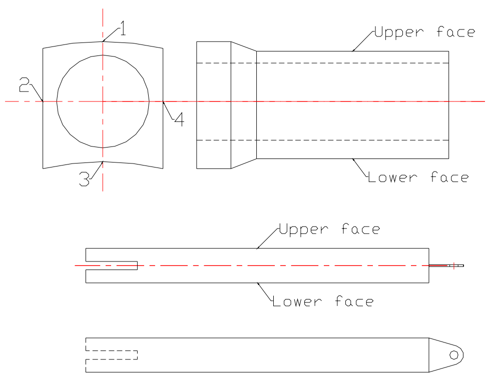

Fig. 13: Measurement positions of the samples.

SAMPLE 1

	Contact Doses Rates (μSv/h)					
Cleaning Time	Point 1	Point 2	Point 3	Point 4	Upper Surface	Lower Surface
T = 0'	3,30	4,20	3,80	4,60	2,50	3,00
T = 30'	1,50	2,40	1,70	2,45	1,70	1,35
T = 60'	1,50	2,00	1,70	2,00	1,30	1,30
T = 120'	1,50	1,30	1,30	1,90	1,30	1,30
T = 180'	1,35	1,30	1,30	1,75	1,30	1,30
T = 360'	1,20	1,30	1,20	1,60	1,20	1,20

	Surface Cont. (Bq/cm²)		Free cont. (Bq/cm²)	
Cleaning Time	Upper Surface	Lower Surface	External Surface	Internal Surface
T = 0'	64,65	66,57	β=1,50 α=0,40	β=1,32 α=0,40
T = 30'	7,09	8,12	Bgd	Bgd
T = 60'	3,11	4,48	Bgd	Bgd
T = 120'	2,39	2,32	Bgd	Bgd
T = 180'	2,22	2,15	Bgd	Bgd
T = 360'	1,90	1,57	Bgd	Bgd

SAMPLE 2

	Contact Doses Rates (µSv/h)					
Cleaning Time	Point 1	Point 2	Point 3	Point 4	Upper Surface	Lower Surface
T = 0'	5,20	10,20	4,80	5,20	4,00	3,90
T = 30'	1,30	1,80	2,20	1,40	1,90	1,80
T = 60'	1,30	1,80	2,00	1,40	1,90	1,60
T = 120'	1,30	1,80	1,50	1,40	1,20	1,60
T = 180'	1,30	1,60	1,50	1,40	1,20	1,50
T = 360'	1,30	1,60	1,50	1,40	1,20	1,40

	Surface Cont. (Bq/cm²)		Free cont. (Bq/cm²)	
Cleaning Time	Upper Surface	Lower Surface	External Surface	Internal Surface
T = 0'	71,50	72,61	β=3,50 α=1,08	β=79,46 α=23,45
T = 30'	20,71	29,53	Bgd	Bgd
T = 60'	12,52	19,20	Bgd	Bgd
T = 120'	10,25	12,69	Bgd	Bgd
T = 180'	8,47	12,00	Bgd	Bgd
T = 360'	5,65	8,16	Bgd	Bgd

SAMPLE 3

	Contact Doses Rates (µSv/h)					
Cleaning Time	Point 1	Point 2	Point 3	Point 4	Upper Surface	Lower Surface
T = 0'	3,65	2,70	3,00	2,50	2,80	3,00
T = 30'	1,75	2,00	1,60	1,80	1,50	1,60
T = 60'	1,60	1,30	1,60	1,70	1,50	1,50
T = 120'	1,50	1,30	1,40	1,70	1,50	1,50
T = 180'	1,50	1,30	1,40	1,60	1,50	1,50
T = 360'	1,10	1,30	1,40	1,40	1,40	1,20

	Surface Cont. (Bq/cm²)		Free cont. (Bq/cm²)	
Cleaning Time	Upper Surface	Lower Surface	External Surface	Internal Surface
T = 0'	73,62	76,25	β=2,00 α=0,42	β=10,24 α=1,67
T = 30'	12,41	12,62	Bgd	Bgd
T = 60'	5,93	7,99	Bgd	Bgd
T = 120'	4,80	5,42	Bgd	Bgd
T = 180'	3,77	4,29	Bgd	Bgd
T = 360'	3,05	4,01	Bgd	Bgd

The following photo (Fig. 14) show a sample before and after the cleaning process. See also Figure 15.

Fig. 14: A sample before and after the process.

SAMPLE 4

	Contact Doses Rates (µSv/h)	
Cleaning Time	Upper Surface	Lower Surface
T = 0'	2,5	2,7
T = 30'	1,5	1,5
T = 60'	1,5	0,8
T = 120'	1,5	1,5
T = 180'	1,5	1,5
T = 360'	1,3	1,3

	Surface Cont. (Bq/cm^2)		Free cont. (Bq/cm^2)
Cleaning Time	Upper Surface	Lower Surface	Exterior Surface
T = 0'	7,16	7,72	β = 8.49 α = 1.98
T = 30'	1,56	0,92	Bgd
T = 60'	0,93	0,54	Bgd
T = 120'	1,11	1,11	Bgd
T = 180'	1,11	0,92	Bgd
T = 360'	0,61	0,55	Bgd

Fig. 15: All samples after 6 hours of decontamination.

6.2 – Inside Measurements of Sample 1

A

Fig. 16: Cut of sample 1.

After the last cleaning, process sample number 1 was cut as shown in Figure 16 in order to measure the contamination and the contact dose rate inside the surface.

The measures at point **A** in Figure 16 were:

Contact Dose Rate	Internal Surface Cont.
1,1µSv/h	3,41 Bq/cm²

7 – Secondary Waste Generation

Secondary waste was collected from tests performed in the most severe condition in order to determine the maximum secondary waste production rate. As assessed by the supplier, the machine was tested with no other load but abrasive media. Under that condition, chips crash against each other continuously instead of against the smoother surface. Therefore, the highest secondary waste production condition is reached.

In the previously described conditions and with the liquid media flux (1liter per hour), the industrial machine produced 1.53 kg of dry solid waste per hour and the laboratory machine, also in the same conditions, produced 0.26 kg of dry solid waste per hour. This means that after 6 hours we have 1.56 kg of dry solid waste (Fig. 17a) from the laboratory machine.

Fig. 17a: Dry solid waste.

The following graphics (Figs. 17b-17d) show the abrasive media average weight loss for the industrial machine during 0, 8, 16, 24 and 32 working hours, the abrasive media samples measures and the standard deviation.

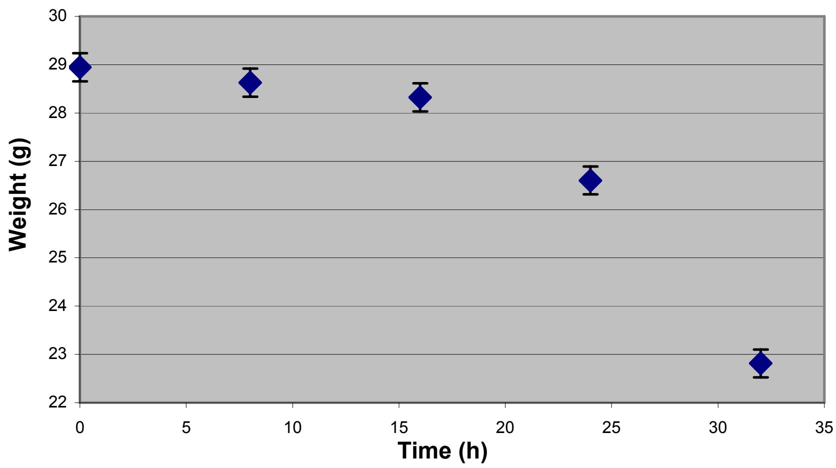

Fig. 17b: Average Abrasive Media Weight Loss

After 32 hours the abrasive media samples average consumption was approximately 6.133 g

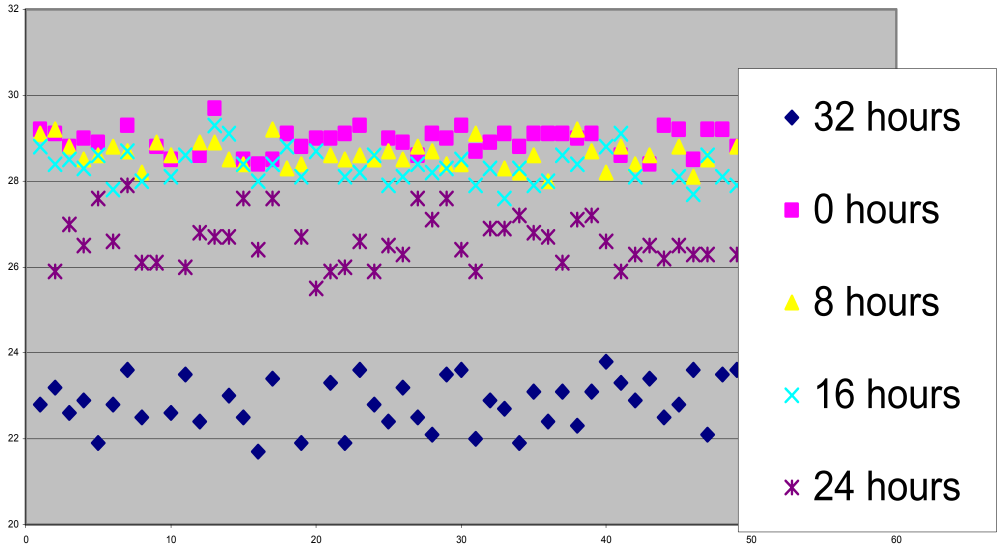

Fig. 17c: Abrasive Media Comsumption.

Fig. 17d: Abrasive media samples from 0 until 8 working hours.

8 – Interactions with other CRP members

- Jin Ho Park (KAERI, Decommissioning Dept.)

Visit to KAERI in 2005 and training of one CNEA professional during 2 month in KAERI in 2006.
Objective: exchange information and experience.

- Sergey Mikheykin (Radon Site, Moscow)

Visit to Radon in 2006 and visit to CNEA in 2007.
Objective: possible agreement of collaboration between companies.

- Vladimir Daniska (Decom)

Visit to Decom and Bohunice NPP in Slovakia
Objective: exchange information and experience.

9 – Conclusions

- After 30 minutes a decontamination factor around 10 was achieved for cleaning the surface contamination of these samples.
- Internal surfaces decontamination needs much more cleaning time than the external surface.
- This technology is an effective pretreatment technique for chemical decontamination process.
- Time, vibration intensity, solid media features and liquid media flux are the main process parameters.
- Size, shape, porosity and hardness are the main parameters of the sample to be taken into account.
- The effectiveness of the cleaning process depends of the cleaning time and shape of the components.
- If a suitable recycling/separation system is not in operation during the process, large amounts of secondary liquid waste will be collected.

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THE DECOMMISSIONING OF THE ASTRA-MTR RESEARCH REACTOR FACILITY

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Nuclear Engineering Seibersdorf GmbH – NES, Seibersdorf, Austria

Abstract
On 31st of July 1999 the only Research Reactor (ASTRA) at the premises of Austrian Research Centre Seibersdorf (ARCS) was finally shut down after an operational period of nearly 40 years. Since the decommissioning of the reactor coincided with the work on the IAEA coordinated research project "Innovative and Adaptive Technologies in Decommissioning of Nuclear Facilities" a concluding report of the project was developed. This paper describes the history of the reactor in relation to the site at Seibersdorf, the decisions leading to the permanent shut down and to the decommissioning of the reactor on behalf of the Austrian government. The planning, financing and the strategy of the decommissioning process are provided; references to the legal requirements according to Austrian regulations and the procedures to gain the decommissioning license (environment impact assessment) are given. Radiation protection procedures and results are reflected. The structure of the overall project is discussed as well as the structure of the main tasks e.g. the removing of the fuel, the recovering and the treatment of the intermediate level waste in the vicinity of the core, the handling and conditioning of the neutron exposed graphite and the Beryllium-elements. The dismantling of the 1600 metric tons of biological shield is described from the determination of the dismantling technique to the selection of clearing procedures and the deposition. The dismantling of the pump-room installations, the processing of the contaminated or activated metals, the dismantling of the ventilation system and the radiological clearance of the reactor building are mentioned. Finally, the paper summarizes the timetable, the flow of the materials, analysis in a brief overview the costs of the project and reflects on the intended and the actual re-use of the reactor building and the demolition of the pump-room. The project was planned well in advance. After successful termination, the conclusion has had to be drawn, that even quite detailed planning has had to be continuously reconsidered throughout the project and adjusted to matters unforeseen. Therefore the establishment of data and reflections on obtained results are also included to a certain extend where the author found it to be essential to the understanding of the decision making process.

1. INTRODUCTION

In 1955 the "Österreichische Studiengesellschaft für Atomenergie" (ÖSGAE) was founded with the proposition to provide the country with facilities to partake in nuclear research and to train staff for foreseen nuclear power plants. The process was similar to the founding of other research facilities throughout Europe at more or less the same time. Whereas Austria never intended to develop national concepts for nuclear plants, independent projects e.g. liquid metal cooling circuits or reinforced concrete structures for pressure vessels were developed and international co-operations with organizations such as IAEA, CERN or the OECD High Temperature Reactor Project - Dragon were staffed. Close contacts were kept with the Austrian Universities.

In 1958 federal agreement was reached to construct a 10 MW MTR multi purpose research reactor of the American Machinery and Foundry (AMF) design at a site approximately 30 km southeast of Vienna near the village of Seibersdorf. Together with the reactor, the infrastructure of the research centre like laboratories for physics, chemistry, biology and health physics and a prototype workshop was built. After two years of intensive building, on the 29th of September 1960, the ASTRA reactor reached first criticality. It was operated at a power level of 100 kW until April 1962. In May 1962 the power level was raised to 1 MW. In 1969 the level of power was further increased to 6 MW and three years later with the addition of another two cooling towers to 7 MW. Since January 1975 the reactor had been operating on a maximum power level of 8 MW and in the last decade of 9.5 MW. The average operating period per year since 1966 was in the range of 500 and 900 MWd/year.

MTR type fuel elements were subsequently changed from the original 90% enrichment (HEU) to the 20% U_xSi_y-Al-design (LEU). A maximum thermal neutron flux of 1.0 10^{14} and a fast neutron flux of 0.7 10^{14} n/cm^{2}s was achieved in the centre of the core. Commercial use of the reactor started around 1970 on a small scale with activation analysis and the production of gamma sources for industrial and medical applications.

After a people's referendum held in Austria in November 1978, generally rejecting the use of nuclear power in Austria and preventing the already built nuclear power plant at Zwentendorf from becoming critical, the scientific use of the reactor subsequently decreased. After several modifications the commercial possibilities of the reactor were extended. A production line for radiopharmaceuticals was added and after the removal of most of the beam tubes around 1985 up to five CNC-controlled precision radiation facilities for production of NTD-silicon were developed and installed. A capacity of around five tons of silica ingots (in diameter between 75 and 125 mm - 3 to 5 inches) while operating at approximately 700 MWd/year added a substantial part to the reactor's income. Nevertheless, the income of the reactor only by commercial use hardly exceeded 50% of the total operating costs of roughly €1.300.000 per year.

2. DECISIONS LEADING TO THE PERMANENT SHUT DOWN

In 1994 the management of the reactor was asked in the course of a routine auditing of ARCS by the government audit office (Rechnungshof) to prepare a preliminary concept and estimate costs for a permanent shut down. This lead to the following results:

- Return of the spent fuel (60 elements) to DOE, USA, 1998 at the earliest, under the assumption that DOE would resume its fuel recovery program. The costs of ultimate disposal of the spent fuel was estimated at around €1.300.000 (to be covered by funds reserved)
- Shut down one year ahead of transport of spent fuel
- 50 to 100 tons activated and contaminated materials to be conditioned at estimated costs of €700.000
- 25 years of manpower for the dismantling work estimated at €1.660.000
- Performance of decommissioning employing the qualified reactor staff
- The building could be kept for alternative use
- No major legal problems were anticipated

The total costs of the decommissioning was therefore estimated with €2.360.000.

In 1995, 16 new fuel elements were purchased from the already closed down Saphir-Reactor in Switzerland; the amount of fuel in stock was estimated to be sufficient for operation from 2002 to 2004.

In May 1996, after a moratorium of eight years, DOE decided to continue its program for recovering fuel of American origin from research reactors. The program was to end in May 2009. Fuel elements intended for transport had to be taken out of the core before May 2006.

Austria, with no nuclear power plants to take care of, had no intentions to build a final storage for minor amounts of high level waste. Therefore the latest possible date for a shut down was May 2006.

In 1997 a new management of ARCS decided, because of political and financial reasons, to shut down the reactor permanently at the earliest possible date. A first deadline for the shut down was communicated as 1st of January 1998. Due to commitments and obligations this deadline had to be extended to the 1st of January 1999, giving the users of the reactor opportunities to arrange for alternatives. This additional time was duly used to establish empirical data related to the activation of the liner, the concrete and other major components.

Between April 1998 and April 1999 on behalf of the Austrian government a more comprehensive study, Ref. [1], was prepared to give a clear picture of the possibilities of decommissioning. Meanwhile, due to a general decrease in radioactive wastes which could be conditioned to the waste treatment facility on site as well as the new legal conditions to be applied, there was a substantial rise in the costs. The costs of labour and conditioning were newly evaluated. Also more comprehensive activity inventory was planned and a timetable was drawn taking into account the qualified staff still remaining to do the work. The following assumptions emerged:

- First possible shipping date for the transfer of the actual 54 spent fuel elements was established with DOE in the fall of 2000. The costs of an estimated €1.800.000 were to be covered by reserves accumulated over the years designated for this purpose
- Handling and storage of 160 tons activated and contaminated materials estimated at a cost of €4.000.000 including conditioning and intermediate storage
- 90 man-years for dismantling, now also to include the conditioning of the intermediate and low level waste, establishing the necessary radiological parameters, clearance of the buildings, radiation protection measures and documentation. This was estimated at €9.000.000
- The work had to be performed with the remaining qualified reactor staff but with the option to use external labour when applying specialized techniques
- The project's duration was timed for six years not taking into account unforeseen delays arising throughout the performance of the task

Total costs of the decommissioning were therefore estimated with €13.000.000 covering all expenses with the exception of the costs for the removal and ultimate disposal of the spent fuel, and a reserve allowances for later final storage of the arising radioactive waste.

3. DECIDING ON STRATEGIES FOR DECOMMISSIONING

After the decision to shut down the ASTRA reactor was reached in May 1998, the question on how the decommissioning should be performed appeared: whether it should be realized after a prolonged cooling-down period or in a predominantly rapid way. In the general public opinion environment which prevailed, the decommissioning of a reactor tended to expand over an extremely long period of time. In the IAEA Technical Guidelines, Ref. [2], three decommissioning phases were distinguished, separated by a few months to several decades. Advantages and disadvantages of rapid decommissioning were discussed and compared to each other, Ref. [3].

To perform a safe and environmentally compatible decommissioning, the possible options and required phases for decommissioning and removal of the radioactive components were evaluated in the decommissioning study from 1999, Ref. [1]. To support the decisions at each phase, an estimate of the activity inventory in the various parts of the reactor and the waste volume to be expected was performed. Measurements of various materials were made as far as accessible and numerical evaluations were used where these were not accessible.

Of the possible options, an immediate dismantling to Phase 1 per IAEA Technical Guidelines, Ref. [2], (storage with surveillance), to be followed immediately by continued dismantling to Phase 2 (restricted site use) was identified as the most reasonable and under the auspices optimum choice. The reasons were that the majority of radionuclides possessing either half-lives up to 80 days which decay sufficiently to permit a continuing dismantling after phase 1, or half-lives so long that waiting periods of more than 50 years would be required to substantially reduce exposure levels. Since these later nuclides showed rather low activity levels, the handling of most contaminated and slightly activated components could be performed without much complication at an early stage. Since the re-use of the reactor buildings was established, the project should immediately continue to phase 3 (clearance and re-use of site). Data given in Table 1 represent activity levels after cooling periods from 2 to 3 years, periods considered typically for that between shut down and the start of the removal of the main radioactive components and the demolition of the biological shield.

Table 1. Average activity inventory in relationship to cooling down periods [1]

Material of Components	Nuclides	Half Live Time	Activity GBq after Shut Down	Activity GBq after 2 years	Activity kBq after 80 days	Activity kBq after 2 years
Be-Reflector Elements	H-3	12.3 a	20	18		
	Co-60	5.27 a	1000	770		
Graphite	C-14	5736 a	1	1		
Grid plate	Cr-51	21.7 d	20000	0		
	Co-58	70.8 d	7000	6		
	Co-60	5.27 a	700	540		
	Mn-54	313 d	40	8		
	Fe-55	2.7 a	14000	8380		
	Fe-59	44.6 d	500	0		
NTD-silicon doping facilities	Ni-63	100 a	22800	22800		
	Ni-59	75000 a	200	200		
	Co-58	70.8 d	38000	30		
	Co-60	5.27 a	200	154		
Control rods	Hf-175	70 d	400000	290		
	Hf-181	42.4 d	3000000	20		
	Hf-178	31 a	0.01	0.01		
Alumina-liner	Sc-46	83.3 d	20	0		
Barite concrete	Sc-46	83.3 d			2.2	0
	Mn-54	312.2 d			0.2	0.05
	Fe-59	44.5 d			2.9	0
	Co-60	5272 a			0.3	0.2
	Ba-131	11.5 d			8.7	0
	Ba-133	10.5 a			2.7	2.3
	Eu-152	12.7 a			0.2	0.2

Preliminary evaluations of the activity inventory gave an estimated amount of 320 kg of intermediate level waste, of about 60 metric tons of contaminated low-level radioactive waste and another 100 metric tons of activated low level radioactive waste. The activities were roughly estimated to total 200 TBq in the intermediate level and 6 GBq in low level.

Another important issue with regard to dismantling and demolition was the methods and procedures to be employed to result in a minimal radiation exposure of the employed staff, Ref. [4]. There was a long standing experience with cutting procedures regarding higher radioactive components in which the exposure of the staff never exceeded very low levels.

4. THE FOUNDING OF NUCLEAR ENGINEERING SEIBERSDORF

Another issue was the intention of the Austrian Federal Government to release the Austrian Research Centers Seibersdorf GmbH (ARCS) from responsibilities connected with the nuclear history of the site as well as from other tasks related to radioactivity. In 2003, Nuclear Engineering Seibersdorf GmbH (NES) was established, an independent organization, operating entirely on behalf of the government and also funded entirely by the government.

One of the main topics of NES is the operation of the Radioactive Waste Management Department (RWMD) acting as central facility for the collection, conditioning and intermediate storage of radioactive wastes arising in the country. Secondly, NES is also assigned with the task to prepare for assistance in safe handling of radioactive materials of medical and industrial origin as well as for emergencies in this field.

A specific contract raised by the Federal Ministry of Transport, Innovation and Technology (BMVIT) as the potential owner covers the comprehensive assignments of NES. Part of the contract takes care of the already progressing decommissioning of the ASTRA, with minor alterations to the original plan as defined within the study of 1999.

Apart from the RWMD and the decommissioning of the ASTRA reactor, NES still operates the Hot Cell Laboratories (HZL), assisting in the conditioning of the Intermediate-Level Waste (ILW), arising now mainly from the dismantling of the reactor. The HZL are due to be decommissioned thereafter and the building to be returned to ARCS.

Last but not least, NES is assigned to decommission laboratories and areas within the premises of ARCS, which were used for "hot work" in earlier days of the research centre and are now to be put to new use.

5. PROJECT PLANNING, FINANCING AND LEGAL REQUIREMENTS

Based on the federal study from 1999, Ref. [1], the decommissioning of the reactor was discussed in numerous meetings with governmental experts. Detailled concepts were drawn up and the main tasks were arranged on a time scale as follows:

Phase 0 – Removal of the fuel elements to DOE/Savannah River Plant until end of 2000
Phase 1 – Removal of the intermediate level wastes by middle of 2002
Phase 2 – Removal of low level wastes to be finished by middle of 2005
Phase 3 – Clearing of the buildings until the end of 2005

The work was divided into preliminary efforts, actual undertaking of the work, the establishment of radiological data, Ref. [5], conditioning and documentation. All tasks were drawn against available manpower.

Since decommissioning work could be performed within the closed containment of the reactor building with negative-pressure, and ventilation and drainage fully in operation, sufficient safety standards could be guaranteed. Virtually, no possibility for a release of activity to the environment during the whole decommissioning process would exist.

In November 1999 the project was finally presented for legislation and duly legalised. The financial envelope of the estimated total of €13.000.000, divided into six equal parts over the years 2000 to 2005 was granted.

According to Austrian legislation, Ref. [6], nuclear facilities operate under federal law, while decommissioning comes under the surveillance of the competent state governments, subject to an environmental impact assessment (EIA). It was therefore decided, that work in phase 0 and phase 1 should be covered by the operating license of the reactor, providing time to prepare for the EIA, essential to continue work in phase 2 and finally in phase 3.

With specific definitions missing it was determined that work until the "final drainage of the primary water" was subject to surveillance under the operating license. This was officially proclaimed in a document under the reference RU4-U-078/000 and dated January 2001. The document resulting from the EIA, two years later, would carry the identification RU4-U-078/047.

Prior to the start of decommissioning, Euratom was informed according to Article 37 of the treaty, Ref. [7, 8]. In a statement received in December 2001 no objections to the decommissioning plans were noted.

Due to further intervention of customers, including the IAEA, with operating facilities on site, the ASTRA reactor was permanently shut down on 31st of July 1999 after 39 years of successful operation. Cleaning and pre-documentation work started immediately thereafter, but no substantial work was performed. The reactor still remained in operating conditions. After a formal guarantee to finance the project over the period of six years was extended by the government late in December 1999, actual decommissioning work began in January 2000.

6. WORKING TEAM, RADIATION PROTECTION

As already explained in Chapter 2, the decommissioning of ASTRA was planned based on the employment of remaining reactor staff. Nevertheless, several companies with experience in the field of decommissioning (e.g. Babcock Noell) were contacted to perform certain tasks like the dismantling of the biological shield, where no techniques were available. Since no sufficient and reliable data could be made available at the time, no binding offers could be gained. Another reason to employ former reactor team members was in view of further decommissioning projects to be performed on site (e.g. Hot Cell Laboratories). It was even found essential, to attract younger people to join the decommissioning crew, to let them gain experience and to continue further extensive decommissioning projects like the total clearing of the site scheduled over the years to come. Contracts were also extended to already retired staff members to establish historical facts and data.

Therefore, a team consisting of 8 former reactor crew staff members (which could be temporarily reinforced by co-workers if demand should arise) was set-up to perform the decommissioning. Another important decision was the employment by the NES radiation protection officer of an independent radiation protection crew under supervision of ARCS, following November 2004. All team members had completed the radiation protection training and had practical backgrounds in operating the reactor, in the handling of radioactive samples and were specialized in specific tasks like operating the reactor's ventilation and cooling system. They were also able to perform at least basic radiological surveillance. Periodic radiation protection briefings supplemented the state of training in this subject.

Between February 2004 and November 2005, 4 persons from the company BBS – Beton-Bohr-Service GmbH, commissioned for the concrete cutting, joined the group. A further 3 skilled workers from the Seibersdorf main workshop, Lindeberg GmbH, joined the team towards the end of the project on a more or less permanent basis. In all, 58 man-years went into the dismantling work. Another 25 man-years were applied to radiological survey and safety, amounting to a total of 83 man-years with 23 people fulltime or part time involved. After the completion of the project only 2 members of the original team still remained.

Radiological surveys, the preparation and evaluation of samples as well as the collection and evaluation of radiological data, the determination of nuclide vectors and procedures based on the obtained data were carried out with the project's own staff and equipment. Only alpha- and low-level-beta determinations had to be commissioned externally. Tasks related to any safety aspect had to be approved in advance by specialists nominated by the competent authorities. Additionally, for matters involving radioactivity (especially the clearance procedures), an independent expert approved by the Competent Authorities covered the progress.

The results of the regular yearly medical examinations indicated no influence of the work related to decommissioning. Regular control measurements at monthly intervals on a whole-body-counter gave no cause to alter extensively established working procedures. The readings of the personal dosimeters over the entire project amounted to a total of 85.6 mSv, averaging 1.07 mSv per year and person. The maximum dose rate encountered for one single person was of 11.2 mSv over the period 1999/2006 amounting to approximately 8% of the theoretical maximum permissible dose of 120 mSV for the same time. Table 2 gives an overview of the doses encountered in relation to the theoretical maximum permissible dose rate over the entire time from August 1999 to December 2006, taking also the transition phase in 1999 before the official start of the project in January 2000, into consideration.

Table 2. Overview doses encountered against theoretical maximum permissible doses

Company	[months]	Theorethical maximum permissible Dose [mSv]	Dose encountered[1)] [mSv]	Dose relative to maximum permissible
NES	842	1403	75.8	5.7
Lindeberg	91	153	5.8	3.8
BBS	69	116	4.0	3.5
Total:	**1002**	**1670**	**85.6**	**5.1**

1) Total accumulated external and internal dose, work in transition periode included

It is possible to deduce from the table, that radiation protection measures in NES were adequate and seriously regarded by all members of the team. An extensive internal paper, Ref. [9], covering the details of the radiological survey was prepared for the project's final report.

7. DESCRIPTION OF THE MAIN TASKS IN DECOMMISSIONING

7.1 Removal and ultimate disposal of the fuel elements

Spent fuel was delivered to the US Department of Energy - DOE in several shipments over the operational time of the ASTRA reactor. Close relationship was, therefore, maintained throughout the years. So Austria's intentions for a permanent shut down of the ASTRA reactor planned for the year 2000 was already communicated to DOE in 1994, before the USA decided to continue their program for recovering spent fuel of American origin from research reactors in May 1996. The following steps were undertaken in order to raise a contract for shipment and ultimate disposal of the spent fuel:

- June 1997: First inspection of fuel element conditions by DOE initiated by the ASTRA management
- November 1998: Visit by DOE, official statement by ARCS about permanent shut down in 1999
- December 1998: ARCS formally applies to DOE about its intention to ship spent fuel
- May 1999: Contract raised by DOE received in ARCS
- November 1999: Return of the contract signed by ARCS, Austrian Federal Government and Euratom

Parallel to the negotiations with DOE, the necessary technical documentation covering the fuel elements to be shipped was prepared by the management of the reactor. A complete set was sent to DOE in November 1999. After some comments by DOE were received in April 2000 the completed papers were returned within the following two weeks. In order to meet the DOE specifications, leak-proving of the elements was carried out in the reactor pool between November 1999 and February 2000. Beforehand the fuel elements were shortened by mechanical cutting to remove the rather bulky aluminium bottom parts, also reducing the price of conditioning by DOE by approximately 10%.
Since the capacity of the crane in the reactor was limited to 10 metric tons, it was decided to carry out the loading of the transfer-flasks within the premises of the Hot-Cell Laboratories (HZL) rather than risking the (alternatively proposed) open-air dry-loading option in front of the reactor building. The decision involved additional extensive preparations like the increasing of the 25 ton crane capacity temporarily to 30 tons and the special design of lifting gear, essential to cope with shipment flasks of the latest design. The spent fuel elements had to be singularly transferred to the pool in the vicinity of the HZL using the NES-transfer flask.

The determination of a suitable transfer flask was one of the most complex efforts of the whole task. Used for shipment of used flasks in the past, the NCS-Goslar and the TN-Pegase, were not usable due to the introduction of new technical and legal standards established in the 1990's. After an international invitation to tender, three suitable offers were received. Evaluation favoured Transnuleaire (TN). In February 2000, became clear that TN would not be able to obtain the necessary

American licensing for their proposed TN-MTR flask in time for a DOE established shipping date of September 2000. After hectic discussions of alternatives, TN assigned transport to Rotterdam to be carried out by Sommer+Grottke/Germany using two NAC-LWT-6 flasks. From Rotterdam to Savannah River transport was the responsibility of NAC. Necessary federal and Euratom permits for the transfer of the fuel and for the international transports and transport insurances were obtained just in time. The 54 spent MTR-fuel elements (310.5 kg of HLW) left the Seibersdorf site on the 31^{st} of May 2001 - six months later than scheduled - and were received at US-DOE Savannah River Plant at the 1^{st} of July 2001. The entire process was documented in detail in Ref. [10].

Ten new elements still remaining out of the purchase from the Saphir in 1995 were sold, again after obtaining the necessary permits, to GKSS-Forschungszentrum Geesthacht, Germany. The transfer took place in February 2003.

7.2 Recovering and treatment of intermediate level activity materials from the vicinity of the core

In immediate succession to the transfer of the spent fuel and still under the operating license, all experimental facilities and components of the reactor within the vicinity of the core or in intermediate storage within the building, e.g. old beam-tube-inserts, were conditioned. Three GNS-Mosaik containers were filled, entirely under water (Fig. 1), dried and placed into intermediate storage with NES Radioactive Waste Management Department RWMD.

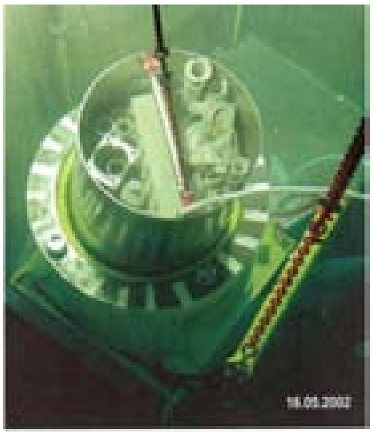

Fig. 1. Under-water loading of a GNS-Mosaik container.

Fig. 2. Clearing the reactor building, transfer of a 21-ton lead cell.

In the course of this custom-designed procedure, remote-controlled equipment had to be built. Required developments and design, technical drawings and part of the production were carried out directly by the decommissioning crew, familiar and experienced with these procedures from the operational days of the reactor.

The task of clearing the reactor building from remaining experimental equipment, obsolete storage facilities and the transfer of the structures of the industrial source services, including a 21-ton-lead-cell (Fig. 2) to NES' Hot Cell Laboratories, were accomplished to 90% under this phase.

During the performance of this task, 492 kg of ILW and 5212 kg of LLW were removed. ILW was fully conditioned within the premises of the reactor, LLW was pre-conditioned and transferred to RWMD for further treatment. Work under phase 1 at the reactor ceased by May 2003: the conditioning of the high exposed graphites and the Beryllium-elements still to be completed at the Hot-Cell-Laboratories.

7.3 Environmental impact assessment (EIA)

According to the original planning, during 2002, while work under phase 1 was still under way, the environmental impact assessment (EIA) requested to obtain the decommissioning license was prepared, Ref. [11, 12]. There was almost no response during the publication period of the documents. The public hearing was held on December 19th, 2002 followed by a license to decommission on April 8th, 2003.

Responsibility for the decommissioning was transferred from federal government to the government of the state of Lower Austria. Since Austria's Health Physics Law was just undergoing the process of homogenizing with EU regulations, it was decided, that clearance standards according to German regulations should be applied throughout decommissioning. This was later to be amended to the national standards once the Austrian Law was enforced in May 2006.

Especially emphasized in the license to decommission following the EIA were the radiological clearance procedures, Refs. [13–15]. The results of clearance measurements as obtained by operations personnel had to be confirmed by the results of a second set of independent measurements by the accredited testing laboratory on site. The results again were to be verified by checking the measurement data and by randomly sampling and measuring 5% of the material to be released by an independent expert specially approved for this task by the authorities. Provided measurement data did show compliance with the regulated release levels. Competent authorities would issue a release certificate for the materials under investigation finally clearing materials either for reuse, recycling, or final disposal.

Preparations for phase 2 was well under way during phase 1, nevertheless, actual work could only be started after granting the license for decommissioning following the EIA in April 2003. Phase 2 mainly comprised the dismantling of the primary and secondary cooling facilities, the removal of the biological shield and finally the dismantling of the ventilation systems, amounting to roughly 160 tons of LLW. About 1500 tons of materials could be cleared.

7.4 Handling and conditioning of the neutron exposed graphite

Material to be conditioned and stored includes approximately 10 tons of reactor-grade graphite originating from the inner (Fig. 3) and outer thermal columns as well as from old-type reflector elements as used between 1960 and 1970 and moderators from late experiments. The activity of the main thermal-neutron activation product, C-14, in the material was on the order of 1000 Bq/g, with other radio nuclides, e.g. Co-60 and Eu-152, present in trace amounts.

Over the 40 years of reactor operation, some of the graphite had been exposed to an estimated integrated fast-neutron flux of 2.2×10^{21} n/cm^2. Since the temperature of the graphite never exceeded 50°C, annealing of lattice defects did not occur and the accumulation of significant amounts of Wigner energy was to be expected. It was decided to preheat graphite exposed to an estimated integrated fast-neutron flux of 10^{19} n/cm^2 and higher under controlled conditions, Ref. [16]. This work was successfully carried out at the NES' Hot Cell Laboratories. Necessary facilities and installations were designed by the decommissioning crew.

Fig. 3. Inner thermal column, recovering of the graphite.

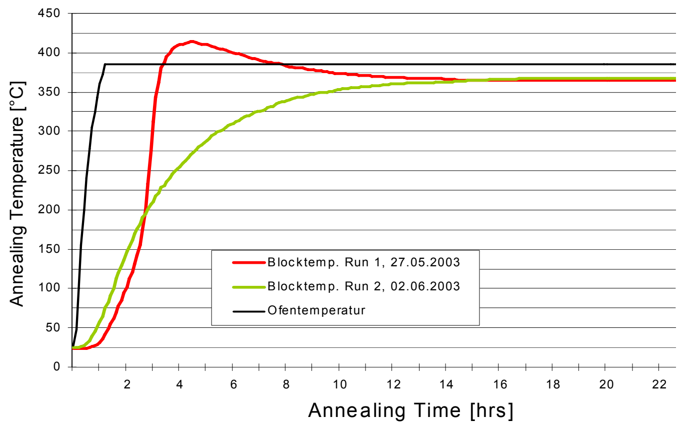

Fig. 4. Temperature behaviour during annealing of graphite exposed to high neutron doses.

Temperature readings (Fig. 4) during the entire annealing process indicated that a considerable release of Wigner energy did, in fact, occur. Therefore, samples of graphite exhibiting a wide range of irradiation histories were available. The amount of Wigner energy, its release kinetics, and associated crystal structure changes were studied as a function of neutron flux – an approach not attempted to date – in a separate project at a later time Ref. [17, 18].

The contents of the outer thermal column, roughly 8 tons of graphite, were removed and conditioned into special stainless steel inserts for the 200-liter standard drum during the summer of 2004. Due to the low levels of exposure to fast neutrons, no preheating was considered necessary. This could be proved by data obtained with preheating graphite from removed areas within the inner thermal column with increasing distance to the reactor core. 2130 kg of graphite from the exterior section could be entirely cleared for re-use.

7.5 Handling and conditioning of the beryllium reflector elements and Hf control rods

Another subject was the conditioning of the ASTRA Beryllium elements. Since no standard procedures were established, it was decided to tightly enclose each of the 25 elements individually into stainless steel containers with a wall-thickness exceeding 5 mm to prevent H3-release.

To cope with the rather intensive radiation of up to approximately 60 GBq of Co-60 in some of the individual Beryllium-elements, storage into two GNS-Mosaik containers was calculated to be sufficient.

The conditioning was again undertaken at the HZL. For this purpose, individual containers for each element were designed from high-grade stainless steel, a remote- controlled orbit welding facility was modified and adapted, and hot cell No. 6 underwent general restoration to handle the work. Two GNS-Mosaik containers had been readied to store the 18 Be-reflector- and the 7 Be-radiation elements together with the active blades of the ASTRA Hf-control rods. The Be-elements and the blades of the control rods were finally stored in sealed condition using the HZL-pool for the transition into the containers.

For permanent safe storage of the total of approximately 3 tons of intermediate level wastes arising through the decommissioning of the ASTRA reactor, altogether five GNS-Mosaik containers were needed.

7.6 Dismantling of the biological shield

7.6.1 Philosophy and preparation

To reach a decision on dismantling techniques to apply to the materials of the activated zone, an extensive sampling program started immediately after the decommissioning license was granted.

To take down the inactive structures of the biological shield (400 m^3 of reinforced Barite-concrete totalling to approximately 1500 tons, several techniques were under discussion, Ref. [19-21]. Finally, dividing the biological shield into blocks of between 7 and 9 tons (limited by the 10-ton-capability of the crane) applying wire cutting techniques was chosen as the most promising method under ASTRA auspices. There were several advantages in preferring wire cutting:

- Measurements and calculations have shown that the risk for spreading contamination due to cutting was almost none existent. Wire cutting needs a lot of water, therefore, no dust would occur. Since an expensive housing was obsolete, a local installation of a HEPA-filtered high power vacuum cleaner, with a cyclone pre-treatment unit to reduce dust and fog particles proved to be sufficient.
- Work could be done with a minimum of manpower. Only two external experts were needed for the handling of the cutting equipment. These were supported by two co-workers and one supervisor of the decommissioning crew, mainly responsible for the controlled gathering of the sludge and the manipulation of the blocks.

- Last but not least, the possibility of applying surface measurements with higher sensitivity compared to the traditional in-barrel technique should guarantee levels of clearance to the standards of re-use.

In order to obtain sufficient data to give a clear picture of the sensitivity of surface measurement, a Canberra ISOCS device was evaluated with positive results. A program for additional internal probing and examining of embedded tubes completed the efforts to prove clearance. The process was presented to, and accepted by, governmental experts. A building directly attached to the reactor was erected to give ample room for clearance measurements and clearance procedures. The ISOCS device was mounted to custom designed gimbals travelling along horizontal and vertical guide rails. All surfaces of the blocks could be reached with a minimum of crane work.

7.6.2 Dismantling the inactive area of the biological shield

Actual cutting started in February 2004. In consideration to the sections in which the biological shield was originally moulded (Fig. 5), a top layer with a vertical extension of 2.4 m was divided into 33 blocks (Fig. 6). The cutting of level 1 was completed on the 16th of March 2004. After removing and clearing the blocks cutting on level 2 with a vertical height of 2.15 m was resumed in June 2004 producing another 43 blocks. By end of September cutting at level 3 with a vertical extension of only 0.94 m (14 blocks) took place. The collection and clearance of the cake was successfully achieved.

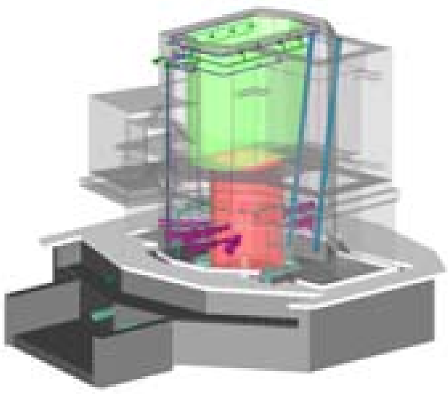

Fig. 5. 3D-study, ASTRA biological shield.

Fig. 6. Removing of blocks 1st layer.

Results obtained by subsequently probing the shield in a vertical pattern allowed for another cut at level 4, 1.8 m below level 3 together with the reduction of the structures of the lower hot cell and the outer thermal column to ground level (level 5). At blocks of the section plane 4, directly adjoining the activated part of the biological shield and crossing over to the activated zone, the compliance with the permissible limit was additionally proven by inspection of core drill samples at prominent regions. Cutting work at level 4/5 ceased at the end of February 2005.

Due to the precautions during the cutting process followed by intensive clearance procedures, all barite concrete blocks could be released to a level sufficient for “buildings for re-use” after minor mechanical treatment. The cleared blocks (1091 tons) were transferred into a storage area specialized in the recovery of building materials (building-remainder-mass-dump). By request of the authorities the blocks were stored in a marked area with the intention of later recycling.

Additionally barite-concrete-sludge was dried sufficiently to ensure safe transport. Clearance data were obtained by sampling at a rate of approximately one sample per 100 litre. Even so, the majority of the amount could be cleared without restrictions from the radiological point of view; it had to be transferred into permanent remainder-mass-dump for technical/chemical reasons.

7.6.3 Dismantling the activated area of the biological shield

Dismantling of the inactive zone of the biological shield began with the dissecting of the activated parts in March 2005.

The bottom part of the biological shield containing the activated zone consisted of rather highly activated concrete facing the side of the former core with activity levels significantly exceeding the levels for release. Still, there had to be also regions with less or no activation, the materials fulfilling the requirements for conventional dumps or even re-use.

From n-flux measurements along the circumference of the pool during the last operations of the reactor together with some activation analysis on barite concrete samples and with some calculations, a more or less homogenous, cylindrical activation depth of 1 meter with a mass of roughly 60 to 70 tons was considered as “activated” zone. In order to reduce the amount of radioactive waste remaining from the biological shield to a minimum, a more accurate definition of the periphery between the zones was necessary, Ref. [22]. Primarily, the “vector” of the radionuclides within the activated concrete had to be determined and with reference to the legal regulations, a “weighting factor” (BF) for release of the concrete had to be developed.

7.6.3.1 Nuclide vector of the barite-concrete and the weighting factors ($BF_{s(Spx)}$)

Samples from different areas of the biological shield were taken and examined via gamma spectrometry, and after chemical processing, additionally by alpha spectrometry and liquid scintillation counting. It became evident that H-3 was dominant. Nevertheless, and because of easy detection by gamma spectrometry, Ba-133 was specified as the reference nuclide.

Table 3. Nuclide-vector and clearance-values (FWSpx) for barite concrete, biological shield

Nuclide	[%]	normalized to Ba-133	$FW_{i(Sp9)}$ [Bq/g]	$FW_{i(Sp5)}$ [Bq/g]
Ba-133	15.9	1	30	1
Co-60	0.9	0.057	4	0.1
Eu-152	1.5	0.094	8	0.2
Eu-154	0.1	0.006	7	0.2
H-3	73.6	4 629	1,000	1,000
Fe-55	9.1	0.572	10,000	200

As soon as the activity concentration of the reference nuclide was known, the activity concentrations of the other radio nuclides could quickly be specified. In Table 3, the percentage fractions of the activity concentration of the radionuclides, present within the barite concrete of the ASTRA biological shield, are indicated in relation to the reference nuclide Ba-133.

According to the German regulations (Dt.StrSchV, annex IV, table 1) clearance value $FW_{i(Sp9)}$ refers to clearance restricted for permanent deposit, whereas $FW_{i(Sp5)}$ refers to clearance for unrestricted re-use.

To decide about release levels with reference to all nuclides present, a weighting factor ($BF_{s(Spx)}$) taking into consideration the activity concentrations and the clearance values of the radio nuclides was determined as the sum of the quotients of the activity concentration (C_i) and the clearance value ($FW_{i(Spx)}$) of the radio nuclides (i) in the nuclide vector of the barite concrete:

$$BF_{s(Spx)} = \Sigma C_i/FW_{i(Spx)}$$

To take into account the values of the normalized nuclide vector (sum of the quotient of the particular normalized value and the clearance value) this weighting factor can also be calculated directly from the activity concentration of the reference nuclide Ba-133. The respective weighting factors were calculated as follows:

Reference nuclide Ba-133: $BF_{s(Sp9)} = 0.065.C_{Ba\text{-}133}$ (materials designated for permanent storage)

$BF_{s(Sp5)} = 2.078.C_{Ba\text{-}133}$ (materials for unrestricted re-use)

Also according to Dt.StrSchV (annex IV, lit. e), further radio nuclides detected or calculated such as Ni-63, Ca-45, Ca-41, Am-241 and Pu-238/239/240 were not considered since the weighting factor of these radio nuclides amount to less than 10% of the total weighting factor ($BF_{others} < 0,1.BF_{total}$). Fortunately, reinforcement steel occurred rather deeply embedded into the barite concrete. Nevertheless nuclide vector and penetration of activation was examined to the same depth as with barite concrete, but there was no need for special considerations within the following assumptions.

If $BF_{s(Sp9)}$ is less than 1, the concrete can be cleared for permanent deposit. If $BF_{s(Sp5)}$ is less than 1, clearance can be granted for unrestricted re-use.

7.6.3.2 Determination of the horizontal activation degree of the biological shield

To determine the horizontal activation profile of the biological shield, 5 horizontal core-drill samples of approximately 5 cm in diameter, and of lengths sometimes exceeding two meters, were taken at different locations along the circumference of the shield. From the cores, and beginning with the most activated side, reference samples with an average length of 5 cm were cut.

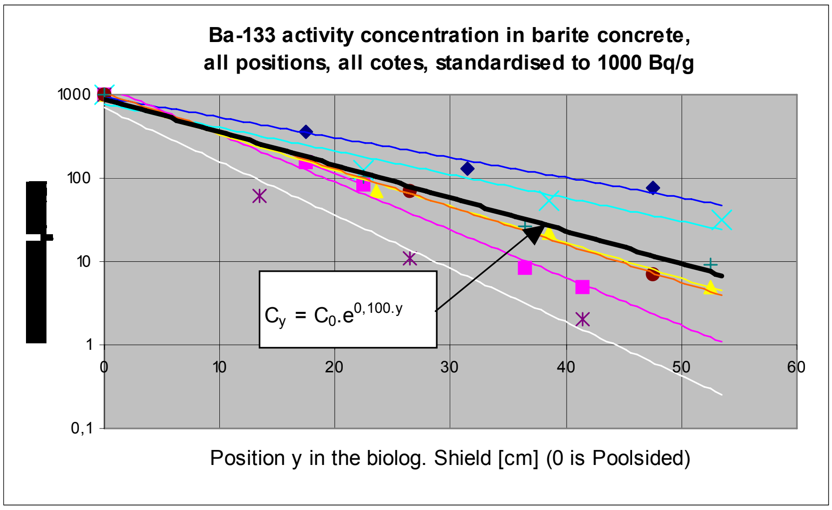

Fig. 7. Horizontal borehole in the biological shield of the ASTRA reactor, exponential decrease of the activity-concentration of Ba-1333, average decrease marked in black.

These samples were examined by gamma spectrometry and the corresponding activity concentration of Ba-133 as reference nuclide [Bq/g] was determined. Using the measurements of each drill core, the horizontal decrease of the activity concentration could be determined. Figure 7 shows the exponential reduction of the activity concentration of all measured distributions at different orientations x around the pool and at different levels z. Since all these drill cores showed comparable reduction behaviors with insignificant differences relative to the penetration depth, easily approximated by an exponential function ($C_{z,y} = C_{z,0}.e^{-ky}$). These exponential reduction values provided an average reduction. Now based on this “average activity-ruler”, with the determined activity concentration of a shallow sample taken from the inside of the pool-wall at a known position x/z it was possible to calculate the accompanying location $y_{(BFSpx = 1)}$ (where activity concentration equals the desired clearance value) for this position within the shield with an accuracy sufficient for practical application.

7.6.3.3 Determination of the vertical activation profiles within the biological shield

Distributions of the activity concentrations at the inner surface of the pool at a distinctive orientation x, at different vertical levels z, were measured by core drilling samples to a depth of 5 cm. Since all these samples were drilled inside the pool, no uncontrolled spreading of contamination occured.

The distribution of the activity concentration found at the inner pool surface could be described dependent on altitude by a normal distribution across height with an expected maximum at the reactor's effective core center: z = +80 cm (the actual vertical center of the fuel-zone being at level z = +90 cm, due to control-rod positions usually at less than 100%, effective n-flux maximum was at z = +80 cm). The Gauss function σ was determined with s = 27 cm (level-difference: 27 cm) (Fig. 8).

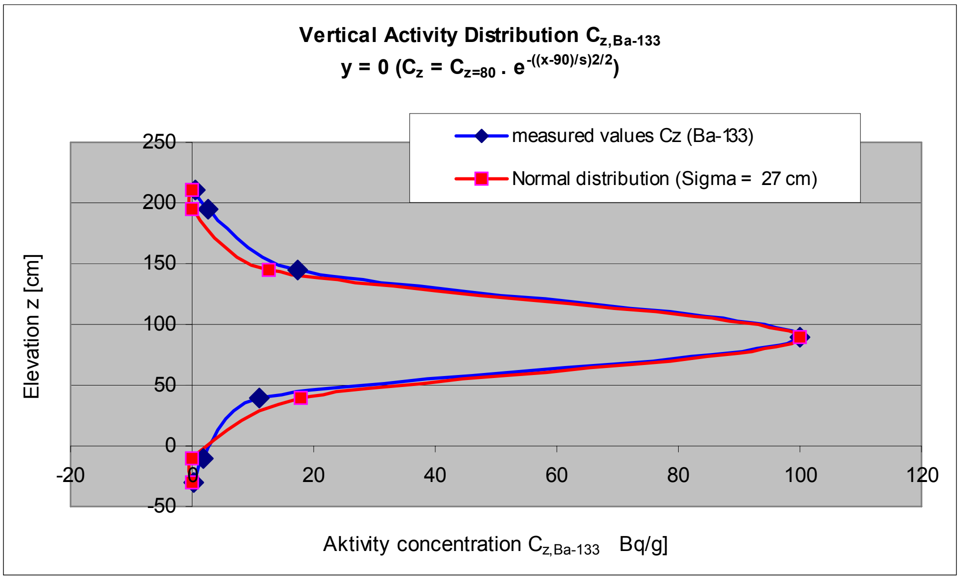

Fig. 8. Vertical distribution of the activity-concentration Ba-133, location y = 0.

Based on this normal distribution, calculation of the vertical distribution of the activity concentration along the inner pool wall by determining the activity concentration at any orientation x and level z = +80 cm was possible.

7.6.3.4 Calculating the horizontal and vertical activation profiles within the biological shield

From all these exponential reductions of the activity concentrations in different orientations and heights of the shield, an average gradient of the exponential function k of 0.100 (± 0.025) was calculated (Fig. 7). On this assumption and on the assumption of the height distribution it was possible to calculate the activity concentration at any location y at a given level z by determining the activity concentration of the barite-concrete at the surface adjoining the pool at the direction x and level z = +80 cm (level of core centre with the highest activation present) (Fig. 8).

Applying the functions described above and by knowing the activity concentrations at the orientations x at z = +80 cm, the locations $y_{(BFs(Spx=1)}$ (clearance values according to Dt.StrSchV equal or less than FW_{Sp5} respectively FW_{Sp9}) could be calculated. Figure 9, left side, positions marked in red, indicate the locations y in the circumference of the pool at level z = +80 cm to determine an activation profile reliable enough to build further actions on. Cores marked in blue were considered necessary at a later stage of dismantling when attempts to drill out beam-tubes failed as described in section 7.6.3.6.

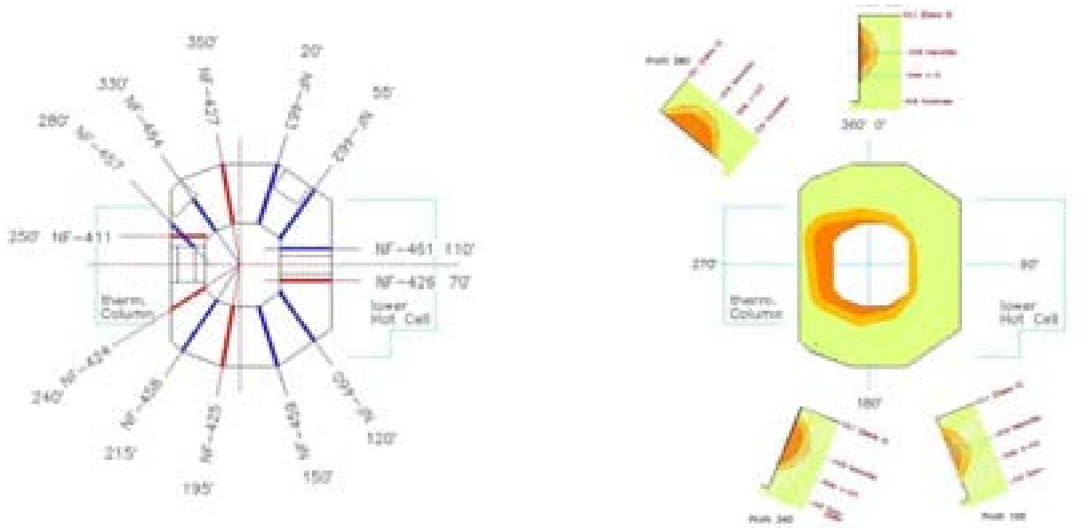

Fig. 9. Horizontal sample positions and corresponding, calculated activation levels.

Figure 9, right side, shows the horizontal activation profile at level z = +80 cm (maximum at core centre height) and several vertical profiles at different angles as calculated by applying the algorithm above. The colour green indicates areas within barite concrete with activities below $BF_{s(Sp5)} < 1$ (unrestricted release), whereas materials activated below release restricted to permanent deposit $BF_{s(Sp9)} < 1$ are indicated in yellow. Only the orange zone is designated radioactive waste.

From a total of approximately 360 tons for the lower biological shield, finally only approximately 25 tons of barite-concrete (including sludge and some debris beyond clearance level) had to be conditioned as radioactive waste. Determining the activation degree by applying the perspectives described above also resulted in considerable reduction of sampling. Since most samples were recovered from the inside of the pool, the spreading of contamination was no real matter of concern. Cross-contaminations of the samples was avoided by recovering samples with expected low activities first.

7.6.3.5 Considering the Reactor's Operational History, Establishing Plausibility

Throughout the dismantling of the ASTRA reactor, findings were compared with the reactor's operational history and the plausibility of the results was established. Decisions in choosing a particular technique, or ways to cope with the various tasks were based on this comparison of information. Experience of retired former staff members and scientists experimenting at the reactor at one time or another was drawn on. Particularly in the case of the examination of the activated area, some of this information proved very useful.

For instance, higher activity levels towards the thermal column were expected. First of all, the core was closer to the pool-wall and the connection between the core via the inner part of the thermal column had to be considered. A thermal shield machined from lead-plates covered the inside of the pool with the exception of the area covered by the thermal column. Finding a maximum between the angles 280° and 330° came as a surprise and was not so easy to explain. Looking into the experimental history of beam-tube E gave some answers.

On the other hand, slightly higher activity levels in direction 180° rather than in the opposite direction 360° came as no surprise at all. After beam-tube experiments ceased in the early 1980's the remaining few experiments were transferred to the beam-tubes E to H. The inserts of A to D as well as J and K were entirely removed. At the side of the core towards 180° as many as 5 silica doping facilities were erected. The neutron-flux was homogenized using cylindrical nickel-shapers around the rotating silica-ingots. To gain the desired accuracy, it was essential to keep the shape of the n-flux over the vertical as constant as possible. Therefore, the reactor was operated with the two of its four control-rods closer to the irradiation rigs constantly drawn at 100%. Reactor-power was regulated entirely with the other two control-rods located further from the irradiation-rigs. Hence the neutron flux at the side 180° during the last 15 years of reactor-operation was on the average higher than towards direction 360°. Horizontal activity distribution around the pool reflected values anticipated.

7.6.3.6 Considerations of possible activation anomalies in the vicinities of the beam-tubes

Some considerations were directed to the detection of activation anomalies in the close vicinity of the ten horizontal beam-tubes, possible due to neutron deflections within former beam-tube-experiments. To minimize risks of cross contaminations probably increasing the amounts of radioactive waste during the procedure of dismantling the outer part of the activated zone, it was decided to remove the entire beam-tube liner via core-drilling by using core-drills with a diameter of 60 cm and of lengths up to more than 2 m (Fig. 10). Closer radiological examinations of the activation profiles along the removed drill-cores were intended. Nevertheless the attempt failed.

Due to the vast amount of reinforcement-steel present at the area (Fig. 11), tangential cuts through steel bars dissecting small, moon-shaped steel-segments (Fig. 12) was unavoidable. The segments, now loosely embedded within the concrete-matrix of the drill-core, immediately caused obstruction of the tool, usually maiming the diamond-impregnated cutting edges too. Time-consuming recovery tasks became necessary. After several similar incidents, while penetrating not more than 0.5 m along the first beam-tube, the whole attempt was terminated.

Fig. 10. Attempted core-drilling at the beam tubes.

Fig. 11. Reduction of blocks dissected at activated zone.

Fig. 12. Moon-shaped segments cut from reinforcement steel.

To gain at least some of the desired data, it was decided to resume horizontal core-drilling of 50 mm diameter samples in a regular pattern at level +0.8 m at the angles between the beam-tubes (Fig. 9, additional samples marked in blue). These were not essential but were useful in confirming the activation profiles already established. No irregularities of any relevance to the intended dismantling were detected.

7.6.3.7 Dismantling and radiological clearance of the outer part of the activated zone

Starting from the location y in the biological shield (clearance location), with a total weighting factor less than 1 ($BF_{s(Spx)} < 1$), the outer areas remaining from the biological shield could be disposed into dumps. With $BF_{s(Sp5)} < 1$ unrestricted release into “building-remainder-mass-dump” was possible, with $BF_{s(Sp9)} < 1$ permanent deposit into conventional “remainder-mass-dumps” was required (compare to section 7.6.2).

After determination of the horizontal and vertical gradients of the activation the cutting sections were set according to the obtained profiles (Fig. 13). As a conservative measure, actual cutting locations were chosen with a 10% safety margin against the calculated borderlines. The barite concrete was cut first along the exemption limit activation profile with reference to Dt.StrSchV, FW_{Sp5}. From these blocks radiological clearance according to Dt.StrSchV, FW_{Sp5} was verified via ISOCS-measurements on all surfaces and additionally via evaluation of various core-drill-samples obtained from prominent positions. With a minimum of re-machining, all blocks could be released without restrictions into a building-remainder-mass-dump.

Determination of cutting edges *Dissecting in progress* *Reduction of the blocks*

Fig. 13. Dismantling the outer part of the activated zone.

The concrete along the activation profile as defined from application of Dt.StrSch V, FW_{Sp9} was cut into blocks. The blocks were reduced to small pieces; the metal parts (10.5 tons of reinforcement steel and the alumina structures from and around the beam-tubes) were separated. The debris was filled into 200-liter-barrels at a rate of approximately three barrels per ton; a total of about 670 barrels had to be handled. The reduction of the blocks was achieved by using an electric-powered, remote-controlled Brock-pneumatic-excavator within a separately ventilated, housed area erected within the reactor building. See Figure 13, picture to the right.

Clearing of the barrels containing barite concrete debris was performed by a Rados RTM640lnc in-barrel measuring system, Ref. [23], available at the company's waste-treatment department RWMD. The RTM640lnc in use is equipped with 10 equal sized plastic scintillation panels arranged around the barrel in a way to reach optimal efficiency. The RTM640lnc-system was calibrated using an specially prepared calibration barrel filled with specially prepared homogenous cutting cake with a nuclide-vector similar to barite-concrete. All barrels were inspected and 95% could be released either according to the clearance values FW_{Sp5} (137 tons), or FW_{Sp9} (101 tons) of the German radiation protection regulation, without further treatment. Due to the results of the RTM640lnc-inspections, the positions of materials with higher activations in barrels with higher activities were known. Approximately 30 barrels were emptied and debris with higher activities manually removed. All barrels containing the reconditioned debris could be cleared in the second run; 8 barrels with selected higher activity debris were deposited together with the active remains of the biological shield. 10.5 tons of recovered reinforcement steel could be entirely cleared for re-use via ISOCS-measurement. The recovered aluminium-structures were treated for re-melting.

After completing the task only the area of the shield with activities clearly exceeding clearance levels remained to be conditioned into the substantially more expensive repository for radioactive waste.

7.6.3.8 Dismantling of the inner activated part of the biological shield (20 tons)

The activated parts of the biological shield with activation levels exceeding the clearance levels FW_{Sp9} comprised about 270 degrees around the circumference of the pool starting from an elevation of 2.1 m above floor level to approximately 0.5 m beyond floor level (the bottom of the pool is at level z = -0.9 m) with thickness to a maximum of 0.8 m (Figs. 14 and 15).

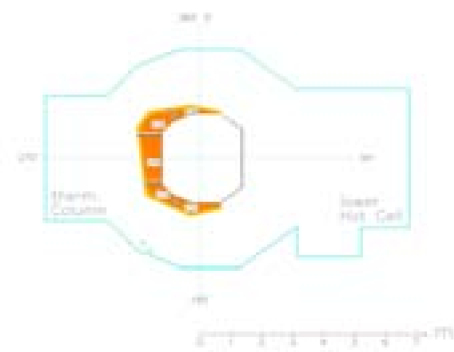

Fig. 14. Determination of cutting edges.

Fig. 15. Biological shield; Activated zone remaining.

Fig. 16. Storing the blocks into Konrag type 2 container.

Taking careful precautions to sustain the sludge, smaller blocks were cut and loaded into three Konrad Type-II steel-containers (Fig. 16). Together with the remaining contents of the 8 barrels of debris (manually separated during the Rados clearance procedures), 20 tons were placed into three Konrad Type-II containers. All together and including approximately 5 tons of sludge from dissecting the lower part of the biological shield, pre-conditioned into barrels, 25 tons of activated materials had to be declared as radioactive waste and were transferred to the Radioactive Waste Management Department RWMD within Nuclear Engineering Seibersdorf GmbH.

7.6.3.9 Removing of the primary tubing embedded within the foundations of the shield

After the removal of the biological shield to ground level, respectively to the bottom of the pool, still segments of the primary cooling system (inlet and outlet, 30 to 35 cm diameter aluminium tubes) as well as parts of primary auxiliary circuits (e.g. overflow, emergency cooling, 7.5 and 10 cm diameter) remained embedded in the foundations of the biological shield (Fig. 17, left). In order to remove the tubing with rather low contaminations on the inside, wire cutting techniques were applied as well.

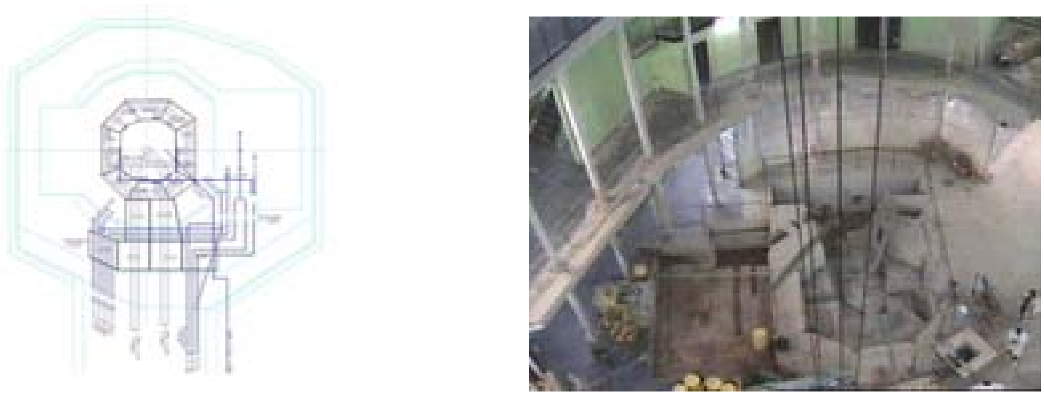

Fig. 17. Removing of primary tubing embedded within foundations of the biological shield.

Down to a level of -2.1 m beyond floor level, concrete containing primary circuit tubing was dissected (Fig. 17, right) and the metal structures were removed. The concrete was either cleared via ISOCS measurements on the surfaces of remaining blocks or via in-barrel measurements as described in the chapters above. The regained metals were prepared and treated for re-melting.

7.6.3.10 Disposing of the materials removed from the biological shield, statistics

Table 4. Statistic of materials removed from the ASTRA biological shield

Barite concrete, inactive zone level 1 to level 5 in blocks, unrestricted release	1091 tons
Barite concrete, activated zone level 6 to level 8 in blocks, unrestr. release	100 tons
Barite concrete, activated zone level 6 to level 8, debris, unrestricted release	137 tons
Reinforcement steel, activated zone level 6 to level 8, unrestricted release	10 tons
Barite concrete, activated zone level 6 to level 8, debris, restricted release	101 tons
Normal concrete, sectional floors etc., unrestricted release	92 tons
Concrete sludge, sufficiently dry for transport, restricted release	36 tons
Barite concrete, activated zone level 6 to 8, blocks and debris, rad. waste	20 tons
Concrete sludge, activated zone, free water removed, radioactive waste	5 tons
Total of materials removed	**1592 tons**

7.7 The processing of contaminated and activated metals

Parallel to the removal of the biological shield, the dismantling of the primary and secondary water installations in the pump room was initiated. The pump room is situated separately from the reactor in a two-storey underground building. Therefore, prior to dismantling, preparations to the building were necessary, which were completed by the end of March 2004. Additionally, economical methods for cleaning and for radiological identification of the metals to be removed had to be developed.

Removing the electrical installations in the pump room took place during May 2004. Thereafter, at the first stage, the potentially inactive components of the secondary water systems were removed. At a second stage, initiated during June 2004, dismantling of the structures of the primary water systems took place. In order to reduce the amount of estimated 60 tons of slightly contaminated metals, it was determined that introducing re-melting procedures would be the most economical way. Since the amount of material would not justify the development of local facilities, contacts with potential European bidders were initiated. Finally a contract with the German company Siempelkamp was raised.

Nuclide fingerprinting of the components of the primary water system was thoroughly done, where Co-60 was established as the reference nuclide as expected. As a surprise, a very low amount of alpha-contamination was detected and identified as Cm-242. After some considerations, it could be attributed to minor contents of natural uranium on the ppm-scale within aluminium structures and the beryllium reflector elements. Due to the long exposure time in direct contact with the fuel elements, activation and subsequent decay to Cm-242 was possible.

7.8 The dismantling of the ASTRA-ventilation system

The installations for fresh-air supply and cross-ventilation of the reactor building were located in the area between upper- and intermediate floor. Exhaust air was led via independent ventilation conducts from defined areas of the building, (e.g. pool-surface, thermal column, hot cells) into the filter and exhaust units located in three rooms attached to the reactor building. The full system was kept in operation until the dismantling of the biological shield, the cleaning and the following radiological surveys pre-decontaminations of the building were completed.

Based on the data from the local and continuous radiological surveillance of the exhaust air during reactor operation, no major contaminations were to expect. This was confirmed by a sampling program initiated end of November 2005. Smear tests established levels not exceeding five times background values. In January 2006, dismantling of the cross-ventilation system was started at the conduits just below the rails of the crane roughly 20 m above floor level.

The segments of the conducts were fabricated out of steel reenforced asbestos-cement plates with the surfaces sealed by decont-paint. This proved very valuable since only simple decontamination by high-pressure water jet was sufficient to clean the surfaces well beyond the levels for unrestricted clearance. Since possible additional hazards due to the contents of asbestos were involved, a safety evaluation was initiated. No further precautions than the already adequate protections for the handling of contaminated equipment had to be enforced.

By the use of a mobile lifting platform and supported by the crane, the segments were dissected, covered with plastic-foil, transferred to a room suitable for wet decontamination within the area of our neighbourhood interim storage and conditioning plant, cleaned and finally cleared without restrictions via in-situ ISOCS measurement. For chemical and technical reasons a permanent conventional disposal for the parts containing asbestos was required. The dismantling of the conducts for fresh- and cross-air was completed in March 2006.

Work was continued, after treatment and disposal of the materials removed, with the dismantling of the metal conducts, the permanent and emergency filters, the blowers and the cooling and heating registers in the rooms of the ventilation system within the reactor building in May 2006, and ceased in July 2007. Finally, during August 2006 the thoroughfares into the attached rooms of the exhaust-air systems at ground-floor-level were dismounted. All together, about 16 tons of materials were treated in the process of the dismantling the ventilation system in the reactor building.

Since work on the ventilation systems started at the upper-floor-level of the reactor building, gradually continuing towards ground-floor-level, and also because of the minor contaminations encountered within the air-conducts, work on radiological clearance of the building could be started in parallel, in May 2006.

8. RADIOLOGICAL CLEARANCE OF THE REACTOR-BUILDING

To obtain radiological clearance of the reactor building, compliance with the release limits according to Austrian Radiation Protection Ordinance had to be proved to the regulatory body. There, in general, the limits for unrestricted release are defined as a maximum dose rate of 10 µSv effective for an individual person per year. Since the structures of the building were never in the effective range of neutron radiation, only contamination due to contact with radioactive materials was to be expected. Usually, buildings would be measured by in-situ gamma spectroscopic devices, a Canberra ISOCS was available and already tried successfully to clear the surfaces of the blocks cut from the biological shield. To obtain results with sensitivity sufficient to prove unrestricted clearance, areas to be measured had to be limited to about 1 m^2 at counting times of around 1000 seconds.

To examine the extensive surfaces of the building (in the range of 2500 m^2), the process turned out to be time consuming and with limited flexibility where decontamination was involved. Therefore, a system of direct measurements, using large-area contamination monitor (beta-gamma-detector BERTHOLD LB165) was chosen. Allowing 10 seconds for the stabilising of the indication, 1 m^2 could be covered within roughly 50 seconds. In restricted areas and to localize contamination detected by the LB165, hand-held monitors (BERTHOLD LB124) were used. In certain cases, the results were referenced by indirect measurements, e.g. smear tests, evaluated on ultra-low-level-alpha-beta-counters (e.g. PROTEAN MPC9604).

Threshold values for the detectors LB165 and LB124 were established taking into account an already defined nuclide vector, the natural background of the concrete and by applying the usual summation formula. For conservative measures and to cope with minor variations in the nuclide vector, threshold values for the actual readings were limited to 25% of the calculated values for unrestricted clearance.

The procedures were described in two working-instructions which were positively accepted by the authorities prior to application.

After the building was cleared from remaining debris and properly cleaned, the wall and floor-surfaces, starting at the upper-floor-level, were divided into marked and numbered areas correlated with the area of the large-area contamination monitor. Documentation was initiated to follow the readings on each area, describing, if necessary, also the decontamination process until clearance levels were obtained.

Radiological clearance throughout the reactor building was initiated in May 2006 and was successfully finished by October 2006. Work was started at the top-floor level and gradually continued downwards until the ground floor was cleared. Decontamination followed immediately after detection of activity and the extensive documentation was drawn up parallel to the progress of the work. The following table gives an overview of the number of measurements, the covered area and the number of contaminations removed.

Table 5. Number of measurements and necessary decontaminations

	Number of measurements	Covered area [m2]	Non-abrasive decontaminations	Abrasive decontaminations
Top-floor, floor area	1070	216	2	2
Top floor, wall area	1440	290	4	7
Intermediate floor, floor area	1836	370	0	7
Intermediate floor, wall area	3180	641	29	23
Staircase, floor area	436	88	0	0
Staircase, wall area	581	117	4	0
Ground floor, floor area	2300	464	0	66
Ground floor, wall area	1537	310	0	17
TOTAL:	**12380**	**2496**	**39**	**122**

With the statement RU4-U-78/091 from October 11th 2006, the unrestricted clearing of the reactor building was officially recognized.

9. SUMMARIZING THE DECOMMISSIONING OF THE ASTRA

9.1 Timetable

The timetable of the project was based on the original planning according to an overall study for the decommissioning of the ASTRA-Reactor from 1999. Over a period of six years, the removal of the fuel, the dismantling of the reactor, the decontamination of the remaining structures (the reactor building), the conditioning of the radioactive waste and the disposal of the conventional materials as well as all matters related with health physics and radiological survey had to be covered. After financial support was finally granted by the end of December 1999, the project officially started in January 2000.

Unforeseen delays were inflicted due to: the process of disposing the fuel (7 months), the comment on the plan according to article 37, Euratom (3 months), followed by the issuing of the decommissioning license (4 months) and administrative problems while erecting the building for clearance measurements (4 months). Those were compensated for by the project management by arranging parallel work using external workers on some of the tasks. Finally, work on the project was completed in October 2006 with the formal acceptance of the cleared building by the authorities 10 months behind schedule. The project was officially terminated by the end of 2006.

9.2 Materials management

One of the intentions of the project management was to minimize the waste, especially where expensive to-dispose-of radioactive materials were involved, but also including conventional waste, where unrestricted clearance and reusability had first priority. The major achievements of the project management in the reduction of radioactive waste were the accomplishment of melting for the very low contaminated metals (roughly 60 tons) and the successful characterisation of the activated areas within the biological shield with a reduction of the estimated 60 to 70 tons to a final 25 tons.

	Mass Flow Phase 1: dismantling of reactor components under operating license
80	inactive, unrestricted, materials for re-use (cleared by NES-Decommissioning-Project)
11	inactive, unrestricted, materials for re-use (cleared by Decont-Services, NES interim storage)
42	inactive, metals, cleared by smelting
7	inactive, restricted, materials into conventional mass-dump
3	ILW, metals, activated, conditioned into 5 Mosaik-containers
9	LLW, metals, activated/contaminated, conditioned into 1 Konrad-Type-II container
7	LLW, graphite, activated, conditioned into 1 Konrad-Type-II container
30	LLW, solid, not burnable, pre conditioned into 100-liter-drums
3	LLW, burnable, pre conditioned into 100-liter-drums
2	LLW, ionexchanger resins, burnable, pre-conditioned into 50-liter plastic-drums
4	LLW, earth, contaminated, pre conditioned inte 100-liter-drums
0	LLW, liquid, not burnable, 122 liters
198	**tons**

	Mass Flow Phase 2: dismantling the biological shield under decommissioning license	
1430	inactive, unrestricted, concrete for re-use	89.8 %
137	inactive, restricted, concrete rubble and sludges, into conventional mass-dump	8.6 %
25	LLW, concrete, conditioned into 3 Konrad-Type-II containers	1.6 %
1592	**tons**	**100.0 %**

	Total Mass, remaining structures within cleared reactor building included
198	active/inactive, dismantling of reactor components (work in phase 1 under operating license)
1592	active/inactive, dismantling the biological shield (work in phase 2 under decom. license)
384	inactive, unrestricted, dismantling remaining structures within cleared reactor building (Oct. to Dec. 2006)
2174	**tons Total**

	Total Mass removed until 31.12.2006, Ways of Disposal	
3	ILW, intermediate level radioactive waste, NES interim storage	0.1 %
80	LLW, low level radioactive waste, NES interim storage	3.7 %
144	materials into conventional mass-dump	6.6 %
1947	materials for unrestricted re-use	89.6 %
2174	**tons Total**	**100.0 %**

Fig. 18. Mass flow dismantling the ASTRA-reactor.

Developing and applying different techniques to establish clearance of uncontaminated materials and taking initiatives to find new applications for still usable materials and equipment and the introduction of the re-melting process for contaminated metals was rewarded with a rather high percentage in unrestricted cleared and re-used equipment. The strategy is reflected in the tables (Fig. 18), where the amount of materials removed is accounted for under different auspices.

9.3 Cost Analysis

The decommissioning project was financed in six equal yearly allocations of €2.180.000 according to the contract from December 1999, amounting to €13.080.000 over the full period. In the contract, it was agreed to re-value the funds over the years following the inflation-index. Further, the long-term storage costs and the costs for the transfer and disposal of the fuel elements were excluded. The costs of the disposal of the fuel to be covered through reserve funds were gathered throughout the years of reactor-operation and set-aside for this purpose.

At the official termination of the project in December 2006, €15.222.960 was credited to the project. Taking into account an average index of 2.5% over the years 2000 to 2005, the amount of €13.080.000 had to be re-valued to €14.224.500, so the actual costs differed for €998.500, equal to an increase of 7 % in the project's total costs.

Besides covering externally inflicted delays within the project, different additional tasks were performed, which were not considered in the original planning e.g.:

- €186.000 for the purchase of 5 Mosaik and 3 KFK-Container
- €207.000 to cover additional costs of the fuel-disposal
- €500.000 (50% of HCL-costs, estimated) for the Wigner-conditioning of 1 ton of graphite and the full conditioning of the 25 Beryllium-elements in the Hot Cell Laboratories
- €70.000 for the installation of a new whole-body-monitor
- €17.000 for the purchase of 5 Konrad-Type-II containers
- €164.000 for the erection of the building to perform the clearance measurements (paid from funds received by selling the 10 new fuel-elements, cost-neutral to the project)

Taking this in consideration, the project was €18.000, or 0.1%, above budget and can be judged as being calculated properly and performed within the given limits. The figures (Table 6) in accordance with the SAP-bookkeeping give information about the costs relative to certain categories.

Table 6. Cost analysis dismantling the ASTRA-reactor

Analysis of the costs of decommissioning the ASTRA-Reactor	**EURO**	**%**
Labour (66.8 years of labour, 2000-2006)	5 244 420	34.45
Material	703 500	4.62
Subcontracts (13.3 years of labour, specialists, experts diamond-wire equipment, clearing of blocks etc.)	2 322 810	15.26
Conditioning in NES Intermediate-Storage-Facility	2 790 620	18.33
Conditioning in NES Hot-Cell-Laboratory - HZL	1 009 220	6.63
Common Costs, Administration, Rents etc.	2 549 250	16.75
Further Costs (transport, insurances, travelling etc.)	602 870	3.96
Total	**15 222 690**	**100**

9.4 Documentation, archive

The project was covered by extensive documentation. All operations within NES followed ISO 9000 quality assurance standards. Overall planning on a yearly basis was detailed monthly. Monthly, quarterly and yearly reports and yearly statistics were prepared. Working instructions for radiation protection and for handling and operating sequences were developed.

Apart from standardized data collection following radiation protection, a daily journal covering the undertaken tasks was kept. For instance, in case of positive results obtained by the monthly whole-body counting or by excretion analyses, the tasks responsible could be easily traced.

Precise data were obtained during material and components handling. Each item from the moment of disassembling to conditioned barrel in the intermediate storage or clearance for re-use, recycling or disposal could be followed at all times. An overall number-based identification system was established and was duly extended throughout the process. Via this system all data, for example within the daily journal, the probes and samples, the CAD-drawings, the extensive photo-documentation and the legal clarification documents were interlocking.

Since there is no guarantee that digital copies are still usable/readable after long years of storage (for some items 30 years and more), it was decided to collect important information and originals preferably in hard copy. To accommodate the extensive documentation from the decommissioning period as well as from the operating period of the reactor, a room on the top-floor of the NES administration building was adopted. It was furnished with steel cabinets for long-time preservation of the documents.

The documentations contain on the decommissioning subject:

- complete documentation of the decommissioning process, planning, operating and evaluation
- monthly, quarterly and yearly reports on decommissioning
- technical documentation on decommissioning
- extensive documentation about radiological clearance measurement and materials flow
- collection of working instructions valid for decommissioning
- papers, publications and books released in connection with decommissioning
  (a comprehensive register is included with the chapter literature)

The documentations on the reactor operation subject contain:

- detailed information about the fuel cycle and disposal over the full operating period
- logbooks reactor-operating room and the radiological surveillance
- continuous records on exhaust air and surveillance of the surroundings
- operating handbook and records of continuous survey by the regulators
- theoretical and technical information concerning experiments (REX- and RBS-reports)
- a complete set of technical drawings of the reactor (AMF, SGAE and supplier)
- daily administrative communication and picture documentation during reactor operation
- personal documentation of the former reactor management

10. RE-USE OF THE REACTOR-BUILDING, DEMOLITION OF THE PUMP-ROOM

In the concept for the decommissioning of the ASTRA-reactor from 1999, a re-use of the reactor building as part of the intermediate storage facility on site was conceived, Ref. [24]. Nevertheless, under the authorization from 2006/2007 and considering the already advanced planning, it was decided for safety reasons by the Austrian government as owner, to invest in further storage facilities within the enclosed controlled area rather than indulge in expensive rebuilding of the reactor containment close to, but outside, the controlled area.

After extensive discussions, the now empty reactor containment will be adapted to house inactive and cleared casks and be used for the interim storage of NORM-waste until the legal requirements for a suitable storage facility are cleared. To fulfil this purpose, the ground floor will be renewed and new lighting will be installed until 2008. Still pending a decision of the owner, an enlargement of the entrance door to a height of 4.2 m and an installation of a basic ventilation system are also foreseen. The attached new building for clearance measurements will continue its inherited designation into the future.

For the re-use of the underground pump room, no reasonable economical propositions were put forward. After unrestricted clearance was obtained, it was decided to demolish the structures including the decay- and storage tank and the basins of the cooling towers to at least a level of 0.7 m beyond ground level, to refill the cavities with suitable clean material and level the area to green-field.

Since the task of removing the structures was not included in, and financed through, the original decommissioning project, it was actually carried out under the project for the general radiological decommissioning of the Seibersdorf site, after 8 months of preparation, during November 2007.

11. INTERACTIONS BETWEEN MEMBERS OF THE CRP

Since the expertise and knowledge represented at the IAEA-CRP "Innovative and Adaptive Technologies in Decommissioning of Nuclear Facilities" covered an unusually wide range of interests in this particular working field, the following brief overview reflects personal interactions arising between members of the CRP-group:

Jerome Dadoumont, SCK/CEN, Belgium – dismantling technologies and equipment (e.g. core-drilling, concrete milling), decontamination technologies (e.g. sponge-jet), advanced wire cutting technologies, economic high-power vacuum facilities

Kurt Lauridsen, Danish Decommissioning, Denmark – handling and conditioning of graphite and beryllium, diamond wire cutting of barite-concrete

Grete Rindahl, Institute For Energy Technology, Norway – visualising areas and buildings in the process of clearance measurements

Sergey Mikheykin, VNIINM, Russian Federation – facilities for treatment and clearance of large amounts of contaminated soil and gravel, calibration of large-area contamination monitors for clearance measurements

Vladimir Daniska, DECOM Ltd., Slovakia, – calculation models for the analysis of costs applied to small facilities (e.g. research reactors)

Yuri Lobach, Institute of Nuclear Research of NAS, Ukraine, – actual materials flow and costs in decommissioning, ways to compare projects under different auspices

Martin Cross, UKAEA, United Kingdom, – treatment of n-exposed graphite

Milagros de las Mercedes Salgado Mojena, Centre for Radiation Protection and Hygiene, Cuba, - calibration of large-area contamination monitors for clearance measurements

Jin Ho Park, Korea Atomic Energy Research Institute, Korea, - diamond wire cutting of barite concrete

12. CONCLUSION

The decommissioning of ASTRA was initiated in 1999 after the conditions of transition were accepted, anticipating IAEA recommendations, Ref. [25], released in 2004. The project's final goal was the release of the buildings for re-use and immediate dismantling was chosen to be the optimum strategy in decommissioning. Decommissioning work followed IAEA's recommendations starting with the removal of HLW immediately followed by ILW and LLW until clearance of the buildings was achieved.

Experience and knowledge were presented and shared with the community e.g. AFR, IAEA and through personal contacts throughout the project Ref. [3-6, 13-15, 17, 18, 22-35].

Summarizing the contents of the decommissioning of the ASTRA-reactor on the Seibersdorf site and considering the full period of the project it should be recapitulated, in general, the dismantling works advanced according to plan. The inevitable unexpected was dealt with successfully, usually in the run of the events. Notable delays were caused mainly by external inflictions, which were not under the control of the project management. Finally, the reactor building and the buildings connected to the reactor could be entirely cleared to the standards of unrestricted re-use.

Summarizing, the many single tasks of the project give clear evidence about the manifold administrative and technical challenges to be met until the goal was reached. It is evident, that the successful decommissioning of the reactor in the described manner was based on responsible preparation by the former reactor operating management, continued knowledgably by the management of NES and the decommissioning personal recruited from the operating staff, making use of the understanding of the functions of the installation in combination with familiarity of applied techniques necessary for the safe handling of radioactive material. It was further essential to fully integrate the operative radiological surveys into the activities of the working crew. Last but not least, the cooperative attitude of regulators, experts and consultants was an essential contribution to the positive outcome of the project.

Finally, it should be manifested that the dismantling of the ASTRA-reactor in the 50th year after the founding of the Austrian Research Center Seibersdorf, within the given limits in time and financial resources and under strict observation of the legal and radiological requirements, was performed without any incidents, neither in the sense of personal safety nor in radiological hazard to the environment, and in doing so, continued the successful tradition of 40 years of safe reactor operation to a termination in dignity.

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DECOMMISSIONING OF NUCLEAR FUEL CYCLE FACILITIES IN THE IPEN-CNEN/SP

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Abstract
During the 1970s and 1980s, IPEN built several technological facilities in pilot scale for the Nuclear Fuel Cycle. In the nineties, radical changes in the Brazilian nuclear policy interrupted these activities and caused the pilot plants' shut-down. Nowadays, IPEN has been facing the problem of dismantling and decommissioning of these old Nuclear Fuel Cycle facilities. Besides the interest in the space for the old units, which constitute a valuable resource that could be used for other institutional priorities, there were also issues such as: the need of constant surveillance, possibility of deterioration of equipment and structures, safety and progressive personnel loss from retirements and transfers to other programs. Some facilities demand special attention requiring preliminary removal of retained materials and treatment of the wastes. A fundamental aspect of the dismantling process in IPEN has been the removal of the retained materials and the final disposition of the radioactive wastes generated during the operations. In IPEN, from the point of view of the facilities dismantling and decommissioning, the presence of contamination in the equipment, structures and buildings, although restricted to relatively low activity levels, constituted the most important concern due to the great volume of radioactive wastes generated during the operations. The reduction of the amount (volume) of radioactive waste is of vital importance. At the different phases of the IAEA BRA-12800 (2004-07 period), besides the rising interest in IPEN's facilities and the improvement of the preliminary dismantling plan, we have proposed and accomplished developement studies of innovative/adaptive technological solutions for solving some of the above mentioned retained material and waste volume reduction problems.

1. Introduction

During the 1970s and 1980s, IPEN built several facilities in pilot scale, for several stages of the Nuclear Fuel Cycle. In the nineties, radical changes in the Brazilian nuclear policy interrupted the research activities and the pilot-plants' shut-down. Since there had elapsed more than ten years from the termination of operational activities, it was evident that there would be no "restart" of the R&D and that it would be necessary to find a final solution for the problems posed by the disused facilities. Since then, IPEN has faced the problem of dismantling and decommissioning of their old Nuclear Fuel Cycle facilities. Those facilities have already played their role in technological development and personnel training, with transfer of the technology to institutions entrusted with the scale-up of these units. As stated, most of the pilot plants interrupted their activities more than ten years ago, due to the lack of resources for the continuity of the research. The decommissioning strategy [1] for these old facilities, dedicated to the technology of the Nuclear Fuel Cycle, follows an approach of gradual advancement of dismantlement, as the resources and technical conditions are available. Some facilities have demanded special attention requiring preliminary treatment of retained materials and/or wastes.

The closing of the program implicated, on one hand, in the reduction of the nuclear activities and, on the other hand, in the appearance of new opportunities: in IPEN's participation in new research programs of strategic importance to Brazil. A change was observed in the profile of IPEN, which was predominantly nuclear, covering an extensive field of research mainly in the area of new materials, alternative energies and environment. Besides the space occupied by the old units, which constituted a valuable resource that could be used for other institutional priorities and programs of national extent, there were also concerns for the need of constant surveillance, for the possibility of deterioration of equipment and structures, for safety and for the progressive personnel loss due to retirements and transfers to other programs.

To understand better the facilities' dismantling and decommissioning problem at IPEN, it is important to describe the scene for nuclear energy in Brazil. The Brazilian National Nuclear Energy Commission (CNEN) is a federal entity, reporting to the Ministry of Science and Technology. In comparison with some developed countries, Brazil has a relatively modest and recent nuclear power program since it has one of the largest hydroelectric potentials in the world. Due to the reduced dimensions of the nuclear market in Brazil and to the lack of large nuclear facility shutdown projects in the near future, there are no companies specialized in dismantling and decommissioning. Brazil has only two nuclear power plants in operation: Angra-I, with 657 Mwe (gross electric power) in commercial operation since January 1985; and Angra-II, with 1345 Mwe (gross electric power) in commercial operation since January 2001. Both are located in the Angra dos Reis County – Rio de Janeiro State, near the cities of Sao Paulo, Rio de Janeiro and Belo Horizonte. Still in the early steps of its construction, Angra-III depends on a governmental decision for its completion (work is currently suspended). Nuclear thus represents only about 2% of the total Brazilian electric installed generation capacity of about 94.7 GWe (2003) [2].

One of the most important difficulties associated with the task of dismantling the old Nuclear Fuel Cycle facilities of IPEN has been the lack of qualified personnel in the area of dismantling and decommissioning. Previous experience in decommissioning in the country has been limited to the activities at the Santo Amaro Plant – the USAM (an old facility designed for processing of the monazite sands and operated by INB - Brazilian Nuclear Industries). Until 2000, the only decommissioning experience in Brazil in terms of nuclear facilities was the closure of the USAM. For fifty years, USAM was dedicated to the processing and production of thorium and rare earths from monazite sands originated from the southeast beaches of Brazil between Bahia and Rio de Janeiro states. The plant was installed in a residential area and in a densely populated region of São Paulo, the largest city in South America. The operations ended in 1992. The decommissioning activities occurred between 1993 and 1999. The Public Ministry of São Paulo State, together with INB, had established a deadline for the plant decommissioning, with daily penalties for lack of fulfilment, and the requirement for regular reports about the status of the decommissioning [3].

Since its foundation in 1956, IPEN has played a decisive role in the development of the nuclear science and technology in Brazil. It was created with the main purpose of performing research and development of peaceful applications of nuclear energy. The Institute's recent history has involved a major participation in the technological development for all steps of the nuclear fuel cycle. One example of the important engagement of IPEN in the technological development in the nuclear fuel cycle area is the isotopic enrichment of uranium by ultra-centrifuge, nowadays at the stage of industrial implantation. This significant achievement was performed in cooperation with the Brazilian Navy. Nuclear fuel cycle R&D activities in IPEN addressed uranium purification, hexafluoride conversion, fuel fabrication for research reactors and thorium and zirconium purification, all of which were accomplished in pilot plant scale with most facilities having been built in the 1970s and 1980s. The facilities were used to promote human resource development, scientific research and a better understanding of fuel cycle technologies.

Basically, for the dismantling operations, the main radionuclides of interest are U of natural isotopic composition and thorium-232. From the point of view of decommissioning those facilities, the presence of contamination in the equipment, structures and buildings, although restricted to low activity levels, constituted an important concern. This was due to the great volume of radioactive wastes generated during the operations. The reduction of the amount (volume) of radioactive waste is of vital importance. It should be noted that the capacity for radioactive wastes stockpiling in IPEN has been exhausted. Another fundamental aspect of the dismantling of the disused facilities is the removal of the material retained during operations within the process equipment.

Due to the large waste volume generated in the dismantling operations, and since most of old nuclear fuel cycle facilities are installed in the **Chemical and Environmental Center** (CQMA) area, one of the main concerns and focuses of research and technological development in this Center of IPEN has been the effluent and waste treatment. Despite the development of some special decontamination techniques, the role of the CQMA has never been the treatment of radioactive wastes or decontamination. It should be emphasized that given the reduction of importance of the nuclear program in the institution, there was a corresponding significant reduction of manpower previously engaged in these activities as, for instance, in the decontamination section. This section is of vital importance for the execution of any D&D program. In this context, the reduction of the waste volume has a significant impact in the decommissioning cost and in the amount of material to be stored. For these reasons, several new developments in the mentioned fields were started, some of them as MSc and PhD subjects. Of course, the conclusion of such research exceeds the duration of a CRP.

During the period of the contract IAEA BRA-12800 (2004-2007), besides the growing emphasis on the old fuel cycle facilities and the improvement of the dismantling plan, the research project has been focused on the development of innovative/adaptive technological solutions to treat hazardous materials, and waste volume reduction related to decommissioning activities.

2. Dismantling and Decommissioning Background in the IPEN

In spite of the difficulties mentioned above, some facilities were actually dismantled at IPEN recently even without previous experience, training support or detailed planning. More traditional D&D models/technologies could not be followed because there is insufficient trained personnel for the function. Limited expertise and lack of information and experience at IPEN in the subject provoked a degree of improvisation. Nevertheless, the operations were accomplished following strict radiological and environmental procedures [4].

The IPEN pilot plants were distributed in groups located in different centres spread out in different buildings as follows:

Chemical and Environmental Center - CQMA: ADU Dissolution (Impure Yellow Cake); Uranyl Nitrate Purification; ADU Precipitation; Calcination of ADU to UO_3; Fluidized Bed Denitration (NUH to UO_3); UF_4 Production - Aqueous route; UF_4 Production - Moving Bed Units I and II – Dry route; Thorium Sulphate Dissolution; Thorium Nitrate Purification; CELESTE-I Reprocessing Laboratory.

PROCON (former Conversion Project): Fluorine Production; Uranium Hexafluoride Production; UF_6 Transfer.

Materials Science and Technology Center - CCTM: Dispersion Fuels Fabrication Facilities and UO_2 Fuel Pellets Production Pilot Plant.

In the first phase of the D&D activities, during the period 2000-02, some laboratories and pilot plants were actually dismantled: a Thorium Nitrate Pilot Plant, Thorium Sulfate Dissolution, the UF_4 Production – Aqueous Route Pilot Plant and the Isotopic Characterization Laboratory. But by that time the focus was not exactly decommissioning. Nevertheless, since 2003 the main objectives and priorities have been a associated with an increase in nuclear facility decommissioning priority. A preliminary report was prepared with the basic procedure to be adopted for the fuel cycle facilities dismantlement at IPEN [5]. This increased profile allowed better knowledge to be developed for each installation that should be decommissioned, the establishing of a decommissioning strategy based on the institutional needs, and efforts to fill in the main gaps in terms of lack of appropriate technical knowledge for the decommissioning and to identify the main technical obstacles that would be faced in the facility dismantling [6-9]. The dismantling operations were performed in four phases:

- From 2000 to 2002 the Thorium Sulfate Dissolution and UF_4 Production Pilot Plant - Aqueous Route in the Building 2 of CQMA were dismantled;
- In 2002 and 2003 the ADU Dissolution (Impure Yellow Cake) and Uranyl Nitrate Purification Pilot Plants, in the Building 1 of CQMA were dismantled. In their place has been built new laboratories for the Environmental Program for use without restrictions. (The latter activities have been performed only in 2006 and 2007);

- In 2006, the dismantling of the Uranium Hexafluoride Conversion Pilot Plant occurred. In its place, which was released without restrictions, has been built part of the laboratories for the Fuel Cell Program [10];
- In 2007, the decommissioning of the UO_2 Pellets Fabrication Pilot Plant was being accomplished [11];
- In 2007 the preliminary plans for the decommissioning of the Pilot Plants installed in Building II of CQMA were being prepared [12].

3. Development Activities of the Research Contract Project BRA 12800

Some activities of the project were proposed to develop technical alternatives to give support to the decommissioning activities, in the present and near future. The main problems identified were:

- the removal of the UF_6 retained in piping and equipments of the conversion unit, with emphasis on hydrolysis;
- the treatment of liquid organic wastes containing U and Pu stored in the Hot Cells of the Reprocessing Laboratory through the oxidation in molten salts;
- adaptation of the molten salt reactor in a glove-box;
- the superficial decontamination of painted carbon steel structures by paint stripping with molten salt baths.

3.1. Removal of Uranium Hexafluoride Crystallized in Piping and Equipments

The removal of the $UF_{6(S)}$ from components of the Conversion Pilot Plant is needed for the conclusion of the facility dismantling activities. Former facility operators proposed the alternative of facility restoration. Nevertheless, this alternative did not seem plausible because it is very expensive, under the circumstances of lack of resources and any scheduled future activities in that technological development area.

Uranium and their compounds pose significant chemical and radiological health hazards. Uranium hexafluoride – UF_6 – is a colorless crystalline solid at standard temperature and pressure with a melting point of 64°C. It sublimates at 56.4°C, is highly toxic and reacts violently with water and many organic compounds, such as oils and lubricants. Because of this, systems and equipment used for processing and carrying UF6 must be extremely clean and free of leaks. Despite the aggressive behavious or UF_6, we proposed to verify the technical feasibility of filling the equipment with water and transforming the UF_6 into UO_2F_2, dissolving it in water, to remove the aqueous solution from the equipment, and finally, to precipitate the uranium in the form of sodium diuranate using sodium hydroxide. The method can be applied in limited facility sections and/or isolated equipment in a very safe operation. This proposed process is much cheaper than the facility restoration and it will represent a very good example of an innovative or adaptive technology and a "tailored solution" suitable to solve D&D problems, case by case, with constrained resources. To assist in these activities an extensive bibliographical review was accomplished, as well as meetings with plant operators and visits to the conversion building to evaluate the status of systems, structures and equipment.

UF_6 does not react with oxygen, nitrogen, carbon dioxide or dry air, but it reacts with water or water vapor, including humidity in the air. When UF_6 comes into contact with water, such as water vapor in the air, the UF_6 and water react, forming extremely corrosive hydrogen fluoride – HF – and a compound called uranyl fluoride – UO_2F_2. UO_2F_2 is soluble in water. Figure 1 presents a flow chart of the UF_4 o UF_6 conversion process.

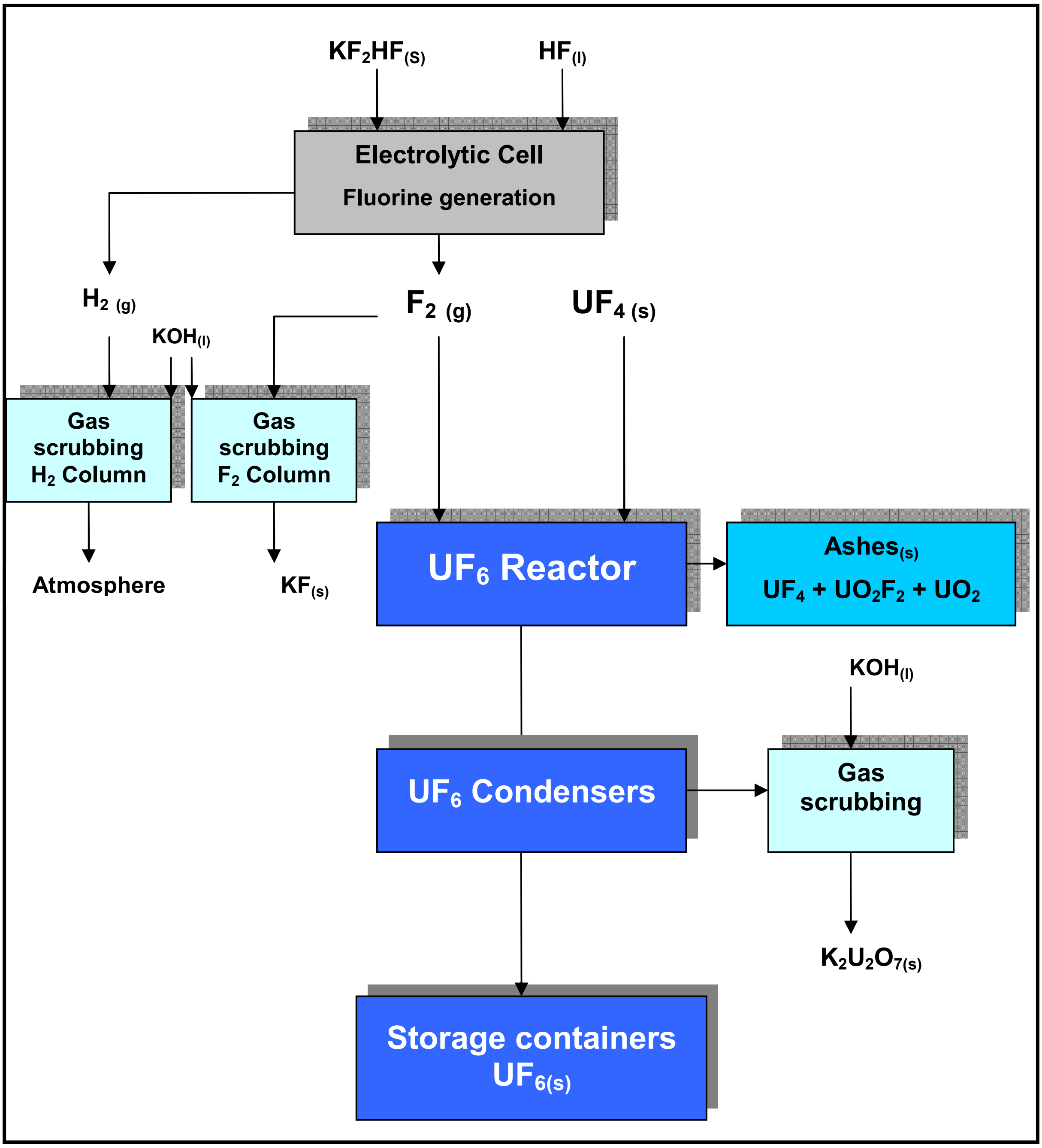

Fig. 1. Uranium hexafluoride - UF_6 – production flow chart.

A risk of the dismantling activities is posed by the presence of solid UF_6 and HF vapor in piping and equipment. Containers and processing equipments that are not leak-tight can release HF formed by the reaction of UF_6 and the moisture present in the air. The risk of particulate material release is increased when the UF_6 is handled in the gaseous state. This reaction can give rise to a subsequent risk of releasing airborne particulate UO_2F_2 in addition to gaseous hydrogen fluoride. The particulate UO_2F_2

tends to settle on surfaces. The UO_2F_2 and HF, which form quickly during a release to the atmosphere, are readily visible as a white cloud (concentration of 1 mg of UO_2F_2 per cubic meter is visible) [13-16].

Aside from nuclear considerations, UF_6 can be safety handled in essentially the same manner as any other corrosive and/or toxic chemical. In Figure 2 is presented a picture of a HF cylinder-emptying operation performed by IPEN personnel. The cylinder-emptying was part of the decommissioning activities. This emptying was necessary because the valve was corroded and HF storage for long periods can result in high pressure inside the cylinder. The HF removed was neutralized with a KOH solution. For the HF handling it is necessary to use special equipment for individual protection, such as plastic coveralls, and masks with independent air supply (the blue rubber hose in the right picture).

Fig. 2. HF cylinder emptying operation and neutralizing of the acid solution.

The variation of density, vapor pressure and physical state with temperature and the chemical and nuclear properties of UF_6 required the development and use of safe handling procedures. The corrosive properties of UF_6 and HF are such that exposure to a severe release can result in skin burns and temporary lung impairment. The inhalation of fumes from very large releases for more than a few breaths may result in temporary lung impairment quite soon after the exposure and, in some instances, kidney damage within a few days. Water-soluble uranium compounds such as UO_2F_2, like most heavy metal compounds, are toxic to the kidneys when inhaled or ingested in large quantities. For uranium of enrichment in ^{235}U bellow 10 %, the chemical toxicity is more important than the radiotoxicity.

An operational problem that occurs frequently in the conversion is the formation of plugs of UO_2F_2 in the piping and valves as a consequence of the humidity entrance. Air leakage into the system must be avoided since the contained moisture will react with UF_6 to produce solid uranyl fluoride, which will plug orifices and foul valves such that they become inoperative. Plugs of UF_6 solid are also possible and localized heating of piping to remove a plug is very hazardous. Liquefying a quantity of UF_6 in a restricted space, such as between valves or two sections with solid UF_6 obstructions can result in rupture of the system and uranium hexafluoride release. Appropriate cooling can be used to control UF_6 releases.

The main problem faced in the conversion facility dismantling is the safe removal of the UF_6 from piping and equipment. The procedure for emptying UF_6 cylinders usually involves transfer by vaporization of the UF_6 using external heating of the cylinder. The heating must be controlled to prevent localized overheating resulting from the low heat transfer characteristics of solid UF_6. Temperatures in excess of 121°C must be avoided. As the isotope composition of the UF_6 eventually retained in the facility corresponds to that of natural uranium, the use of water for removal operations is simpler, as there was no concern about criticality. Cylinders containing residual quantities of UF_6 may be cleaned by water washing.

The use of UF_6 hydrolysis was proposed as a method for cleaning the equipment from the dismantling of the conversion facility since to reactivate the unit would be expensive and complex in the current situation. The hydrolysis reaction in excess of water can be performed by evaporation of the UF_6 that is introduced in a reactor with circulating water to form uranyl fluoride and hydrogen fluoride [14, 15, 17]. In this case, the reaction is described as:

$$UF_6\,(g) + H_2O\,(l) \rightarrow UO_2F_2\,(\text{dissolved}) + 4HF\,(\text{dissolved})$$

For equipment cleaning, the solid UF_6 can react directly with water. For small components, such as limited sections from the piping structure containing crystallized UF_6, the problem is relatively simple because it is possible to open them under water. Nevertheless, equipment containing undetermined amounts of UF_6 requires careful analysis because the reaction of UF_6 with an excess of water is violent. For larger pieces, such as the crystallizers, the restriction of an eventual vapor expansion can be dangerous. Unfortunately, no autoclave is available to perform the controlled sublimation of UF_6, given the dimensions of the equipment. Consequently, the most reliable emptying alternative would be the connection of the crystallizers to a cylinder of nitrogen, pre-heating of the gas and circulating it inside the equipment. As the UF_6 evaporates, it would be bubbled through a container with water. The resulting solution can be neutralized with sodium hydroxide and the uranium recovered as a precipitate of sodium diuranate.

Other alternative is to put the equipment (or the piping section) inside a furnace or other heating system and to assist the process with a vacuum pump connected to the component. A small test-rig was assembled to simulate the removal of $UF_{6(s)}$ from equipment, using a heating system (furnace) and vacuum pump to assist the sublimation. Between the equipment and the vacuum pump is connected a vessel with water. Uranium recovery from evaporated UF_6 follows the process as discribed in the previous paragraph. The experimental equipment assembled can be observed in Figure 3.

Fig. 3. Heating system (left) and vacuum pump with small steel cylinders (right).

3.2. Thermal Decomposition of Organic Solutions Containing U and Pu

The facility CELESTE-I of IPEN is a laboratory where reprocessing studies were accomplished during the 1980s and in the beginning of the 1990s. The last operations performed occurred in 92-93. The research activities gave rise to radioactive wastes in the form of organic and aqueous solutions of different compositions and concentrations. Part of the wastes generated during the operations is stored in a hot cell. The facility is pictured in Figure 4 and a schematic drawing is presented in Figure 5, which shows the waste tanks in position.

Fig. 4. Hot cells of CELESTE-1 laboratory.

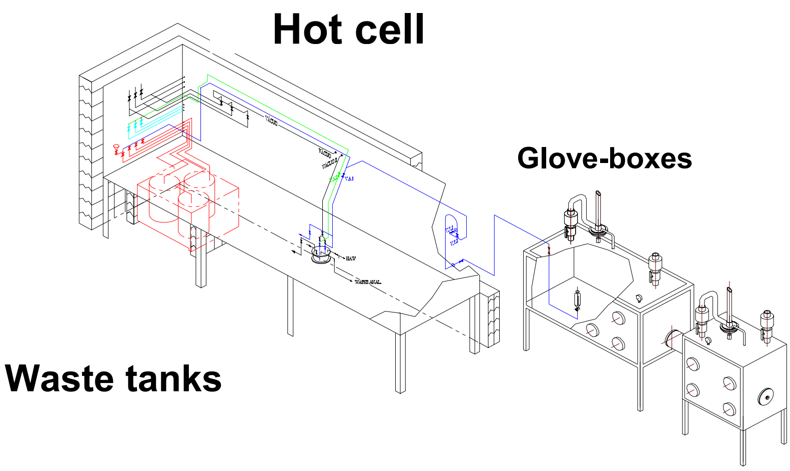

Fig. 5. Schematic drawing of the hot cells and the liquid waste tanks.

Liquid wastes were generated during the process operations accomplished in the CELESTE-I Laboratory. The wastes that were stored in the facility waste tanks are from plutonium retention studies in columns. The wastes contain U, Pu, Np, Am, HNO_3, TBP (tri-n-butil phosphate), hexone and TTA (tenoiltrifluoracetone). Some wastes, from calorimetric and potentiometric analytical procedures, are stored in glass flasks, inside glove-boxes or under hoods. These wastes, containing U, Pu and Am, were used in chromatographic extraction studies. Besides this, in the facility several flasks containing considerable volumes of different liquid wastes from analytical procedures of U determination, containing U, HNO_3, TBP, hexone, pyridine, aluminum and DBM (dibenzoilmetane) are also stored. The volume of organic and aqueous solutions stored in the tanks and flasks in the facility is about 470 liters. As treatment of this kind of liquid wastes is not possible in IPEN, we proposed a study of a waste thermal decomposition process. A study based on the process of submerged oxidation of wastes in a molten salt (sodium carbonate) bath was proposed.

The interest in the decomposition of hazardous wastes by advanced methods as an alternative to incineration, and especially through molten salt oxidation evolved from suggestions by an expert from the International Agency of Atomic Energy - IAEA, Dr. James Navratil (at that time, linked to the Idaho National Engineering and Environmental Laboratory - INEEL / US DOE), in a Technical Visit to IPEN in 1997. In fact, some activities related to this process had already been performed at IPEN before the proposal of the Research Contract. Among several advantages, such as oxidative reactions that transform completely the components of the organic solvent into CO_2 and water alone, the process equipment can be built in small scale compatible with the space available in the hot cells. The process accepts a water content in the organic solution of about 20% without problems. Molten salt oxidation equipment had already been built at IPEN [18] and different organic wastes have been tested (dichloroethane, dichlorodifluoromethane and toluene). In Figure 6, a schematic drawing of the process is presented.

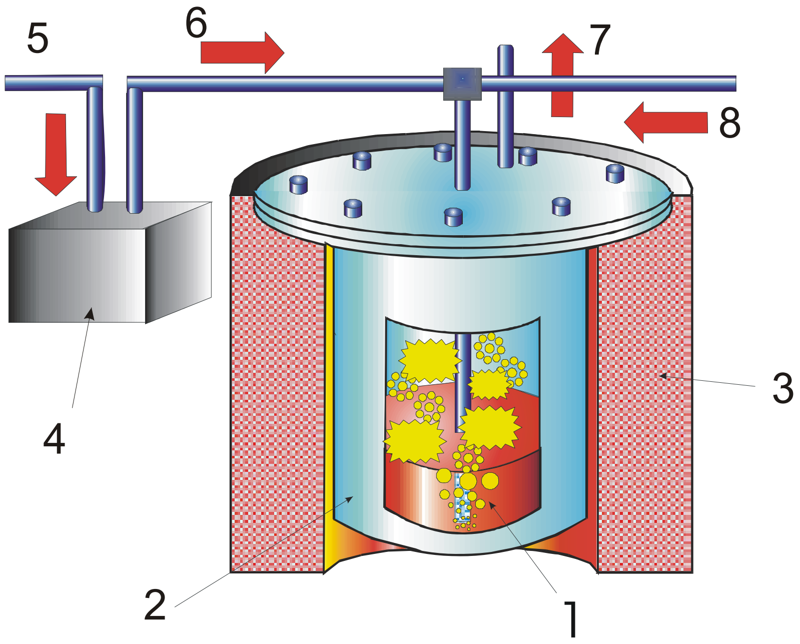

Fig. 6: Schematic drawing of the molten salt oxidation process where 1- molten salt, 2 – reactor vessel, 3- heating system, 4- wastes pressurized reservoir, 5- compressed air, 6- waste feeding piping, 7- Off gas, 8- Air/oxygen injection.

The molten salt, after cooling, can be dissolved in water for uranium and plutonium recovery as a mixture of oxides by filtering (U_3O_8, PuO_2 and oxides of metals such as Ni, Cr and Fe from the reactor vessel corrosion). As the molten salt in the range of temperatures studied (900 to 1020°C) is very corrosive, the reactor vessel must be constructed with an expensive nickel alloy – Inconel™ 600. The developed equipment can be observed in Figure 7. In Figure 8, the salt dissolved in water and a filtering operation for U oxide recovery can be seen. Nevertheless, considering the purposes of the research project and the presence of Pu in the wastes, it was necessary to develop a smaller reactor and heating system, compatible with the dimensions of a single glove-box. In Figure 9, the old and new reactors are both visible.

Fig. 7. Heating system and reactor (left), molten salt (center), reactor removal.

Fig. 8. Filtering to separate the oxides present in the salt and salt crystallization.

Fig. 9. Comparison between the two reactors (left), reactor and its heating system (right).

The smaller reactor was adapted to a glove-box with the objective of evaluating interferences and associated difficulties. Later, the equipment will be set up in a glove-box inside the CELESTE-1 Laboratory. In Figure 10, the assembly of the integrated heating system and reactor in the glove–box can be observed.

Fig. 10. Assembling of equipment inside glove-box.

Some tests were carried out to evaluate the equipment. It was necessary to increase the thermal insulation to avoid damage to the glove-box acrylic walls. Another problem was identified, which occurred as a consequence of the reactor dimensions. We verified that the reactor vessel height must be increased to avoid solidification of the salt in the upper reactor region (flange) where the molten salt was carried by the gases. In this location, it solidified and obstructed the passage of the gases from the waste decomposition, as can be observed in Figure 11. This problem limited the flow of waste and oxidant gases tested during the experiments.

Fig.11. Clogging of salt solidified in the upper region of the reactor (left), molten salt can be observed through the central hole left by the injection lance (right).

Several decomposition tests of different organic wastes have been performed in laboratory equipment developed at IPEN with excellent results. The completeness of the oxidation reactions was evaluated by mass spectrometry of the gases released. A flow chart of the process can be observed in Figure 12. In spite the previously mentioned problems, the process feasibility has been demonstrated. Nevertheless, it will be necessary to introduce some modifications in the equipment to permit its assembly inside a glove-box, such as increasing the reactor height, improving the thermal insulation of the heating system and using a larger glove-box. As the reactor material is expensive, the modifications could not yet be implemented. However, a new glove-box has been selected and prepared, as can be observed in the Figure 13. The tests accomplished will permit to build in the future optimized equipment for decomposition of the U and Pu solutions stored in the hot cell tanks and flasks to meet the needs of the facility dismantling.

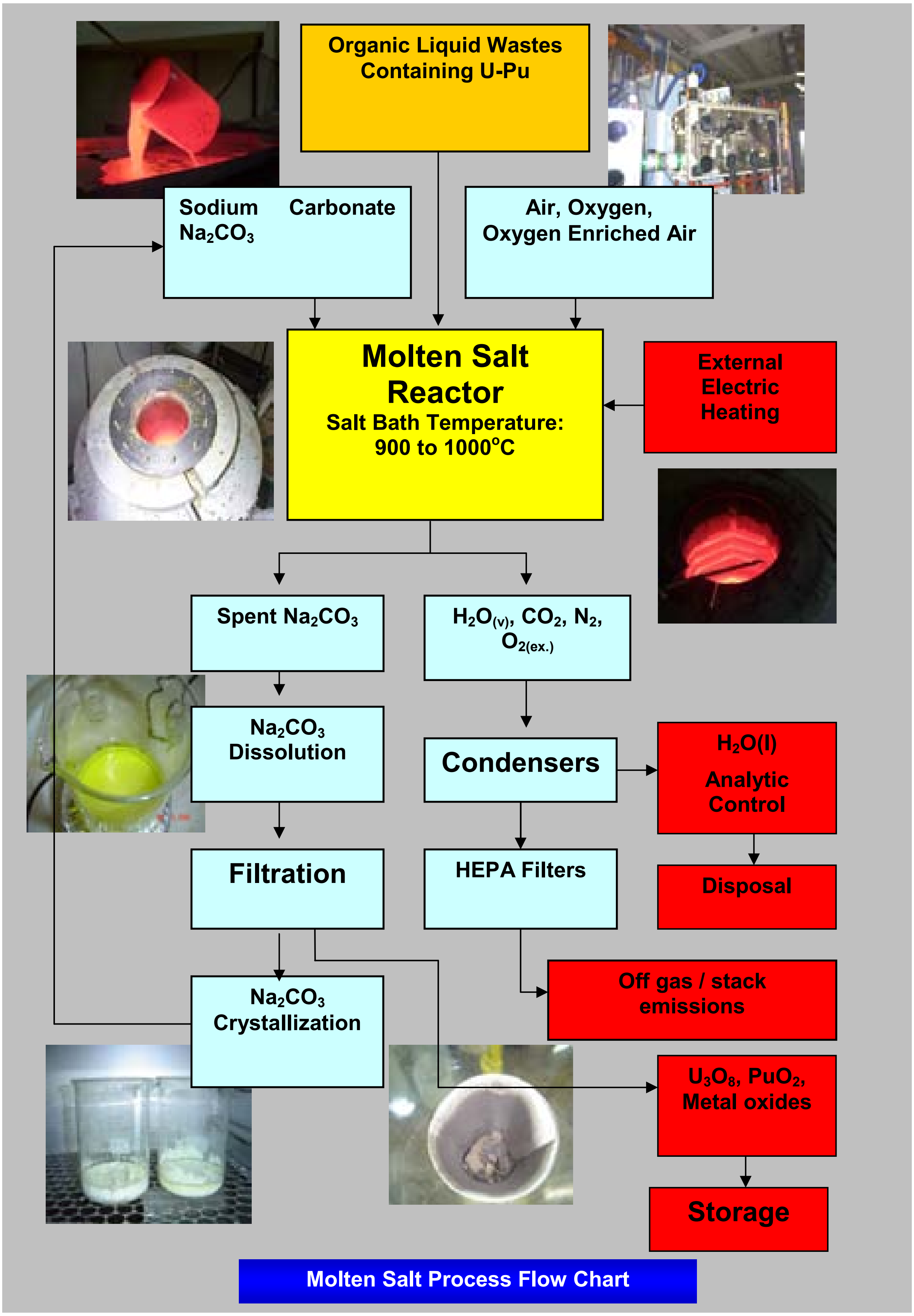

Fig.12. Molten salt oxidation of contaminated liquid wastes flow chart.

Fig. 13. A larger glove-box, previously used in the process (already contaminated) is being prepared for future assembling of the molten salt equipment.

Assembling of the molten salt reactor has to be done within the glove-box. Despite having spent significant time and effort to decontaminate it, decontamination of the selected glove-box (that had been used previously with experiments with plutonium) was not satisfactory, and the radiological protection authority's approval could not be obtained. Finally, it was decided to assemble a new glove-box using the same basic construction design. This decision meant delays and increased expenditures, but was essential, considering the problems encountered with the decontamination of the equipment aimed at its release. The assembly schedule is late. Besides the decision for equipment change, the releasing of budget funds in 2007 was particularly late, occurring only in the end of March. In Figures 14 to 16 are presented the overall assembly concept, the construction of a totally new glove-box and the HEPA filters assembly.

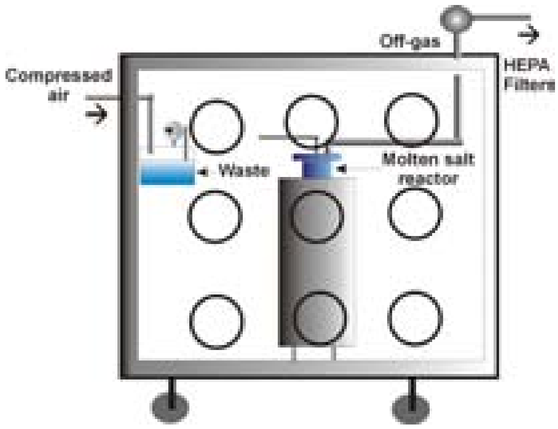

Fig.14. Conceptual assembling of the molten salt process in a glove-box.

Fig.15. Different views of the new glove-box in construction.

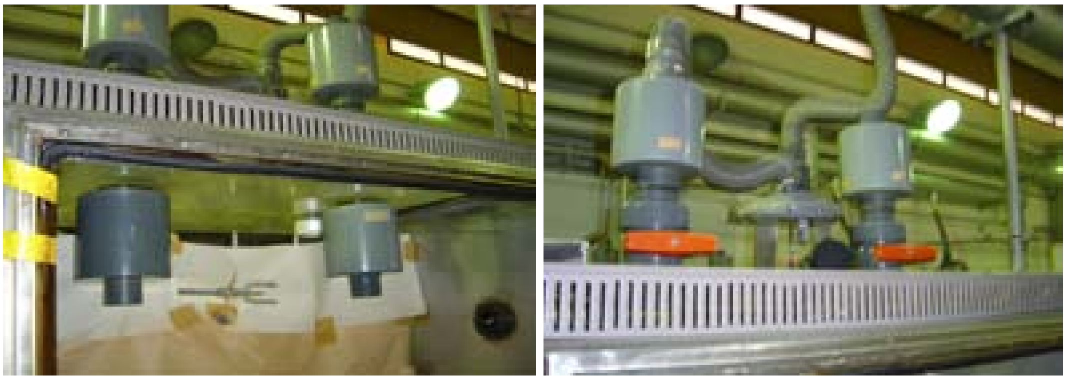

Fig.16. HEPA filters for assembling in the glove-box.

Efficacy of the process was evaluated in terms of extent of organic compound thermal decomposition. Surrogates were used to simulate the wastes in the tests: 1,2-dichloroethane and toluene. Main process parameters included: salt temperature, waste and air flow rates, and lance geometry. A Gas Chromatograph coupled with a Mass Spectrometer was used to analyze molecular fragments present in the off-gas. These were retained in a resin XAD-4 and eluted with n-hexan (Fig. 17). The results can be observed in the Figures 18 and 19. A destruction efficiency of DE = 99,999986% was reached.

Fig.17. Gas sampler and CG/MS analysis.

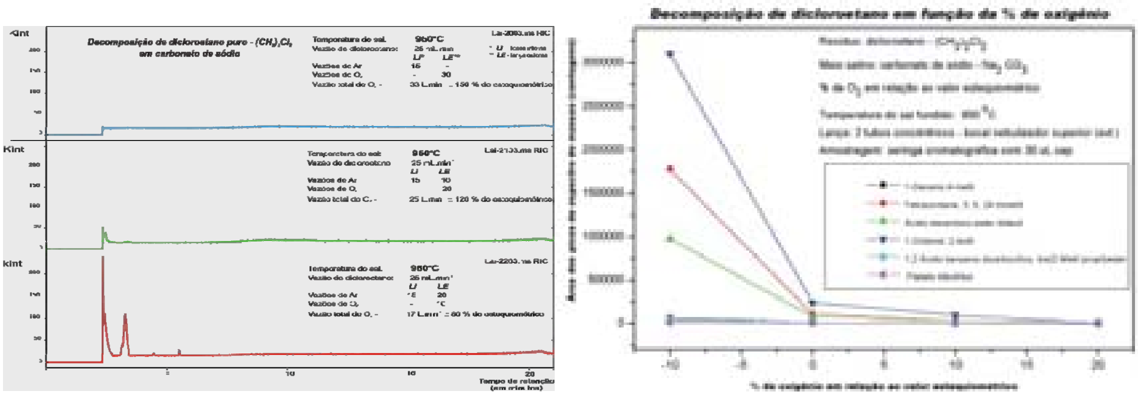

Fig.18. Results of the decomposition of 1,2-Dichloroethane (surrogate).

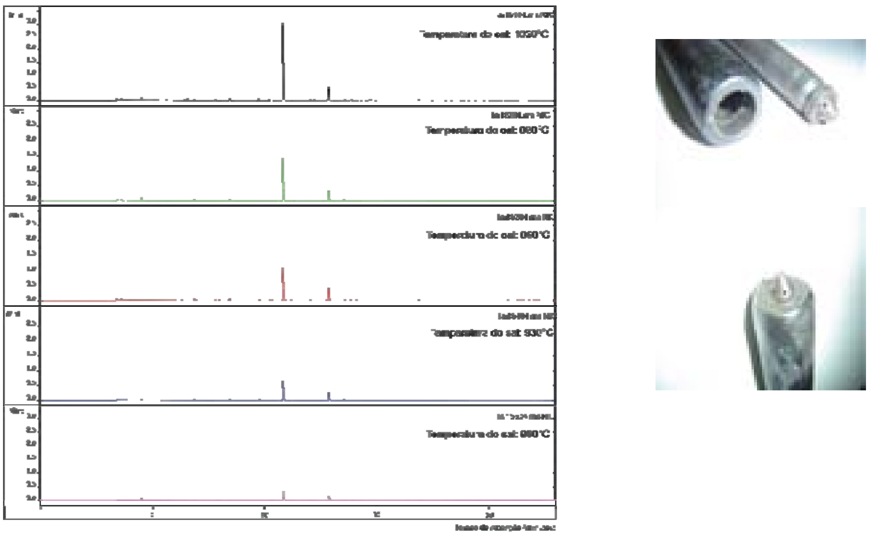

Fig.19. Results of the decomposition of toluene (surrogate; lance geometry).

3.3. *Decontamination of carbon steel structures by stripping in molten salt bath*

The proposal for the fourth phase of the project has been the evaluation of the molten salt bath for the solution to the problem of superficial contamination removal. One problem detected during the nuclear fuel cycle dismantling activities already performed in the IPEN was the difficulty of treatment of contaminated painted carbon steel structures. During the dismantling activities of IPEN´s pilot plants, several tons of contaminated steel structural waste have been generated. Several methods were attempted for superficial contamination removal with reduced contaminated waste volume, such as: decontamination by rinsing methods employing acid and alkali solutions (chemical method) or abrasive removal (physical method). However, the results employing these methods were not satisfactory.

Removal of the combination of paint and carbon steel corrosion creates serious problems in terms of decontamination efficacy and workers exposure. It was proposed to adapt the molten salt system, developed for the decomposition of organic solutions, to evaluate the possibility of using the salt bath as a stripping and descaling system. *A priori*, the organic compounds of the paints would be decomposed, and the steel corrosion products mixed with uranium compounds would be retained in the molten salt as oxides. After dissolution of the salt, filtration can be used for separation of the radioactive wastes (in the oxide form these are insoluble in the salt). The salt can then be crystallized and reused in the process.

One of the objectives of the project is to determine the best salt mixture composition for optimization (minimization) of the bath temperature consistent with efficient removal of paint and corrosion products. It was found that the most contaminated regions in the steel structures were the oxidized or corroded areas. Paint had been applied to reduce the corrosion rate, since the structures had been exposed to aggressive environments. The presence of several layers of paint made the decontamination process much more difficult.

Advantages of the method, as seen above, are that: (a) the oxides could be removed from the salt by dissolution (sodium carbonate, sodium hydroxide or mixtures of both salts) followed by filtration; and (b) the salt solution can be recycled by evaporation and crystallization. The recycled salt can be used several times (it is necessary only to replace that lost in the process). The main parameters to be evaluated are the molten salt bath temperatures and the residence time of the structure pieces. Possibly, lower temperatures can be employed (when compared with the temperatures employed in organic solutions decomposition). Lower temperatures are expected with molten salt baths constituted of sodium carbonate and sodium hydroxide in different proportions. The addition of sodium hydroxide to the sodium carbonate reduces the melting temperature of the salt (in comparison with pure sodium carbonate).

The operational conditions of the fuel cycle pilot plants were deleterious to structural materials, submitting them to an chemically aggressive environment as well as to radioactive liquids and dust that form deposits on the structure surfaces. The chemicals employed, like acids, caused corrosion of the carbon steel. To protect the structures and to increase their useful lives, the structures were painted. During the facility’s operational life, several layers of paint were applied in an attempt to stop or reduce the corrosion rate. In Figure 20 are shown the main carbon steel structures employed in the pilot plants of IPEN (after removal of the internals), and the corrosion status of some structures after more than a decade since interruption of the activities.

Fig.20. Corroded perforated carbon steel structures before and after dismantling.

Some attempts were made to reduce the volume of these wastes. Nevertheless, the only decontamination methods available were chemical methods, such as pickling treatments, employing acid solutions (with nitric or citric acids) and alkaline solutions (sodium hydroxide). Various concentrations of such solutions were tested. Some tanks removed in the dismantling operations were re-deployed in the decontamination process, as can be observed in Figures 21 and 22.

Fig.21. Stainless steel tank adapted for decontamination of components.

Fig.22. Polypropylene tank adapted for decontamination of large steel pieces.

The results obtained with the processes described above were not fully satisfactory. Besides the generation of large volumes of effluents that needed treatment by precipitation (for example, with sodium hydroxide added to U or Th acid solutions) and analysis before final disposal, the decontamination of some components was also impossible. In spite of achieving reasonable efficiency when applied to stainless steel components, those perforated, and the several-times-painted carbon steel structures were not decontaminated sufficiently to be released by this method.

Ultrasonic equipment available was employed to increase the efficiency of decontamination. In Figure 23 it is possible to observe the ultrasonic equipment as well as the decontamination of some components. This equipment permits application of some additional techniques such as solution heating and bath agitation. The acid and alkaline solutions were again employed, varying parameters such as concentration, temperature and time of immersion.

Fig.23. Assembling the ultrasonic equipment.

Some good results obtained with certain materials, e.g. aluminum pickled with sodium hydroxide solutions (even without ultrasonics) and stainless steel- for which the best results were obtained with nitric acid solutions at temperatures between 60-70°C and ultrasonic support. These good results were obatined even for complex shape components. However, the results were frustrating with painted carbon steel parts.

The painted and perforated carbon steel could not be decontaminated satisfactorily and all the waste generated during the dismantling had to be conditioned (by cutting to suitable dimensions) and stored in special drums or steel boxes. Storage in drums is not suitable because a lot of work is necessary to cut the structures into small pieces and the space is not filled completely, as shown in the Figure 24. Storage in steel boxes is much more advantageous, since the cutting work for the components is less and the space is better occupied, as shown in Figure 25. Nevertheless, the boxes are expensive, there is free space inside them and they still a lot of area needed for their storage.

Fig.24. Drums containing parts of carbon steel structures.

Fig.25. Contaminated structural parts of carbon steel stored in steel boxes.

To reduce the large volume of waste generated in the dismantling operations, a variety of methods were tried as described above. For some materials the available methods presented good results. This was the case of stainless steel components, since more than 7000 pieces with different shapes and sizes were effectively decontaminated during the D&D operations of the Uranium Dissolution and Uranium Purification Pilot Plants. Good results were also obtained in the decontamination of aluminum components. The volume of waste in this case is not so significant. Nevertheless, the radioactive waste constituted by painted and corroded carbon steel components deserves special attention, since the volume generated during the facility dismantling is, at least, one order of magnitude bigger than the previously mentioned materials. Unfortunately, with the changes in the Brazilian R&D Nuclear Programme, the decontamination area at IPEN was not properly equipped for the task of treatment of the waste volume that has been generated during the on-going D&D activities. There were insufficient tools and/or facilities to accomplish the needed operations.

Through certain tests and measurements, it was possible to conclude that the radioactive contamination was mainly in the most corroded areas of the components. When paint had been applied over these corroded areas, the acid or alkaline pickling was particularly impaired and the decontamination could not be accomplished satisfactorily. In Figure 26 is shown a piece of a painted carbon steel that was immersed in concentrated sulfuric acid aqueous solution for 3 hours. Half of the piece was submerged and half had no contact with the acid solution. It is possible to observe that the paint almost completely protected the surface. Hence, the removal of the paint and the corroded products from the steel surfaces is an essential pre-conditionning step.

Fig.26. Painted carbon steel submitted to acid pickling (left side) with small effect.

A review of methods for coating removal was carried out. As the main objective of this report is not to discuss details of such methods, just a summary of the important information obtained for the research carried out to fulfill the proposal is presented. In accordance with US EPA - United States Environment Protection Agency,"*a cleaner technology is a source reduction or recycle method applied to eliminate or significantly reduce hazardous waste generation"* [19]. Pollution prevention should emphasize source reduction technologies over recycling, but if source reduction technologies are not available, recycling is a good approach to reducing waste generation. The cleaner technology must reduce the quantity, toxicity, or both, of the waste produced. Also, the cost of applying the new technology relative to the cost of similar technologies needs to be considered.

Several methods have been applied for the minimization of radioactive waste [20-22]. The choice of a coating removal process for radioactive material in the form of carbon steel pieces must take into account, among other factors, that it is not necessary to have a high quality of finishing, since the main objective is the release of the material as iron scrap. This is different from other applications, where the main objective is to recover the component for reworking (appliances industry, for instance). The reduction of waste volume and the consequent reduction of expensive containers and of space for storage are the driving forces. The presence of radionuclides as aerosols should be avoided since the contamination would spread out and the workers need to wear special clothes, respirators and eye protection equipment. Blasting also generates high noise levels. Besides this, pieces with complex shapes need much work. In same cases, such as internal parts (tubes), blasting is not effective.

During the molten salt stripping process, by-products of the reaction of the salt and the coating, as well as the radioactive contaminants present (mainly in the corroded and oxidized areas of the metal surface) accumulate in the bath. Even when the bath is saturated with by-products, stripping will continue.

As molten salt coating removal works by combustion of the coating organics, the molten salt stripping process replaces solvent strippers. The organic content of the coating or paint (hydrocarbons) will be oxidized by reaction with air and the salt bath and will form only CO_2(g) and H_2O(v). In spite of the by-product, used salt has a small volume containing mostly metal oxides and metal salts (formed by reaction of pigments with salt bath materials). The salt can be recycled.

It was decided to explore former experience with molten salts to investigate the possibility of its application as a potential method to solve the problem of the radioactive waste generated during the dismantling operations of the Nuclear Fuel Cycle Pilot Plants of IPEN. This waste is characterized as large amounts of waste in the form of superficially contaminated painted carbon steel structures. Basically, almost all waste generated in the operations is relatively homogeneous in composition;

constituted mainly by perforated steel structural components. On one hand, the perforated components and complex shapes would require a lot of work (material such as solid carbon dioxide, for instance) if blasting is used. On the other hand, as the pieces are made from steel, they can be processed in relatively high temperatures without problems. As additional advantages, immersion of the piece in molten salts avoids the generation of particulate material in suspension and the method can process parts with complex shapes and the internal walls of tubes. In spite of the energy requirements to melt the salt, it is possible to select salt compositions of low melting point and low prices. Also, the salt can be reused after dissolving, filtration (to retain the present radioactive oxides) and recrystallizing.

Regardless of the fact that use of molten salt has already been developed for some industrial coating removal process, references have not been found in the literature about its use for radioactive superficial contamination removal. The molten salt stripping process relies on chemical oxidation of the coating by a molten salt bath. Then, the main questions for this study are (considering the application of the method for radioactive waste treatment):

- It is necessary to develop specially formulated salt compositions that provide very low melting temperatures;
- It is necessary to develop low-cost saline mixtures;
- It is necessary to develop saline mixtures with higher effectiveness in coating and rust and/or corrosion products removal;
- How critical optimizing the contact time between pieces and the salt bath to get the required contamination removal is;
- It is necessary to establish a salt recycling method to reduce as much as possible the generation of secondary wastes.

The system previously built for treatment of radioactive organic solutions by molten salt oxidation was adapted for the first exploratory experiments of coating and contamination removal by stripping with molten salts. The former experience of using molten salt for wastes treatment at IPEN had employed sodium carbonate – Na_2CO_3 - that is cheap and permits the combination with halogens to form halide salts (that are retained in the bath) and release $CO_2(g)$. The first experiments were accomplished with this salt and a nickel alloy reactor. In Figure 27 can be observed an experiment of paint removal with sodium carbonate.

Fig. 27. Molten salt stripping - introduction (left) and removal (right) of the sample.

As showed in Figure 27, the combustion of the paint creates a flame. The temperature of the bath of pure sodium carbonate was 900°C. The disadvantage of this compound is its relatively high melting temperature ~ 852°C. The salt bath provides thermal inertia and effective heat transfer to avoid hot spots or temperature excursions. The molten salt also acts as a gas scrubber which retains the non-volatile reaction products (metal oxides and ashes). As the main functions of the molten salts are to act as a heat transfer medium and catalyst to oxidize the organics in the paint, and the costs of the process and some technical difficulties are associated with higher temperatures, it is important to remove the paint at temperatures as low as possible. Thus it is necessary to look for molten salt formulations that can provide low temperature and efficient paint removal.

In the first set of experiments 42 samples were cut from contaminated structures. Each sample was characterized by measurements of its activity in terms of counts per second. As the activity is not constant along the piece, the maximum counts per second measured was adopted as an indicator. Since the kind of structure (perforated steel in L form), material (carbon steel), dimensions and coating (painted) are always the same, the main process variables were selected: salt composition, salt temperature and residence time. A new heating system and two bigger reactor vessels were built to perform the molten salt stripping tests. The assembly of the equipment inside an exhaust system is presented in Figure 28. Four different salt compositions were selected for the tests: pure sodium carbonate; pure sodium hydroxide; the eutectic mixture of sodium carbonate (41% in mass) and sodium hydroxide (59% in mass); and sodium hydroxide with addition of about 10 % in mass of sodium nitrite (oxidizing salt). Two immersion or residence times were selected: 10 and 20 minutes. Temperatures selected were 450°C and 650°C for the different salt compositions and 900°C for pure sodium carbonate. The amount of salt in the reactor was approximately 3.5 kg, corresponding to a height of about 40 cm.

Following immersion, the samples were removed from the salt bath and rinsed with water for salt removal and cooling. The rinsed items were measured again to determine their respective activities in terms of the maximum counts per second. After this step, the items were submitted to a pickling treatment with sulfuric acid (98%) diluted in water (20% in volume). In Figure 29, it is possible to observe the monitoring of an item and the structures appearance before and after the stripping treatment by molten salt immersion.

Fig. 28. The new equipment assembled inside an exhaust system.

Fig. 29. Monitoring (left), items before (above right) and after stripping (bellow right).

Considering that a possible advantage of the molten salt stripping process is the possibility of reducing the generation of secondary wastes by means of salt recycling, some preliminary tests were accomplished. The salt retains the contamination in the form of metal oxides and ashes that are insoluble. The dissolution of the salt in water followed by filtration permits the removal of those oxides. For dissolution, it is possible to use the same water employed for the rinse step. The remaining aqueous salt solution can be then concentrated in a hot-air stove or crystallizer and the salt can be reused in the process, returning to the molten salt reactor vessel. This way, no liquid waste results from the process. The only waste generated is the small amount of material retained in the filter. In Figure 30 it is possible to observe the radioactive solids retained in filtration, and in Figure 31 the recovering of salt by crystallization. The process flow sheet summarizing the experiments accomplished is showed in Figure 32. The results are summarized in the Table 1, i.e. the different conditions and results for the preliminary experiments. The results are presented before (only with the molten salt stripping treatment) and after the acid pickling treatment.

Fig.30. Filtration of the solution obtained from the salt and the final solid waste.

Fig.31. Crystallization of the filtrate by heating (left) and recovered salt (right).

Table 1. Different conditions and results for the preliminary experiments (BG= 4 counts/sec).

Sample	Salt composition	Temperature of salt – (°C)	Time (minutes)	Counts/s Initial	M. salt*	Final
1	Na_2CO_3	950	10	70	10	4
2	Na_2CO_3	950	10	40	4	4
3	Na_2CO_3	950	10	50	5	5
4	Na_2CO_3	950	20	20	5	4
5	Na_2CO_3	950	20	70	3	4
6	Na_2CO_3	950	20	70	4	4
7	NaOH + Na_2CO_3**	450	10	100	20	4
8	NaOH + Na_2CO_3**	450	10	40	20	5
9	NaOH + Na_2CO_3**	450	10	70	6	4
10	NaOH + Na_2CO_3**	450	20	120	25	5
11	NaOH + Na_2CO_3**	450	20	30	7	4
12	NaOH + Na_2CO_3**	450	20	90	6	4
13	NaOH + Na_2CO_3**	550	10	300	13	4
14	NaOH + Na_2CO_3**	550	10	300	15	15
15	NaOH + Na_2CO_3**	550	10	60	10	5
16	NaOH + Na_2CO_3**	550	20	90	20	5
17	NaOH + Na_2CO_3**	550	20	150	25	10
18	NaOH + Na_2CO_3**	550	20	80	10	7
19	NaOH + Na_2CO_3**	650	10	30	3	5
20	NaOH + Na_2CO_3**	650	10	100	7	4
21	NaOH + Na_2CO_3**	650	10	70	10	5
22	NaOH + Na_2CO_3**	650	20	250	7	5
23	NaOH + Na_2CO_3**	650	20	70	3	4
24	NaOH + Na_2CO_3**	650	20	50	4	5
25	NaOH	650	10	25	5	4
26	NaOH	650	10	90	5	5
27	NaOH	650	10	40	4	4
28	NaOH	650	20	50	5	4
29	NaOH	650	20	70	4	4
30	NaOH	650	20	80	6	5
31	NaOH	450	10	80	5	4
32	NaOH	450	10	40	5	4
33	NaOH	450	10	40	6	5
34	NaOH	450	20	80	5	4
35	NaOH	450	20	30	4	4
36	NaOH	450	20	40	6	4
37	NaOH+ $NaNO_2$***	450	10	50	14	4
38	NaOH+ $NaNO_2$***	450	10	25	5	4
39	NaOH + $NaNO_2$***	450	10	50	9	4
40	NaOH+ $NaNO_2$***	450	20	30	5	4
41	NaOH+ $NaNO_2$***	450	20	70	4	4
42	NaOH+ $NaNO_2$***	450	20	30	7	5

* Results before (only with the molten salt stripping treatment) and after the acid pickling.
** 59 wt% of NaOH and 49 wt% Na_2CO_3 *** 90 wt% of NaOH and 10 wt% of $NaNO_2$.

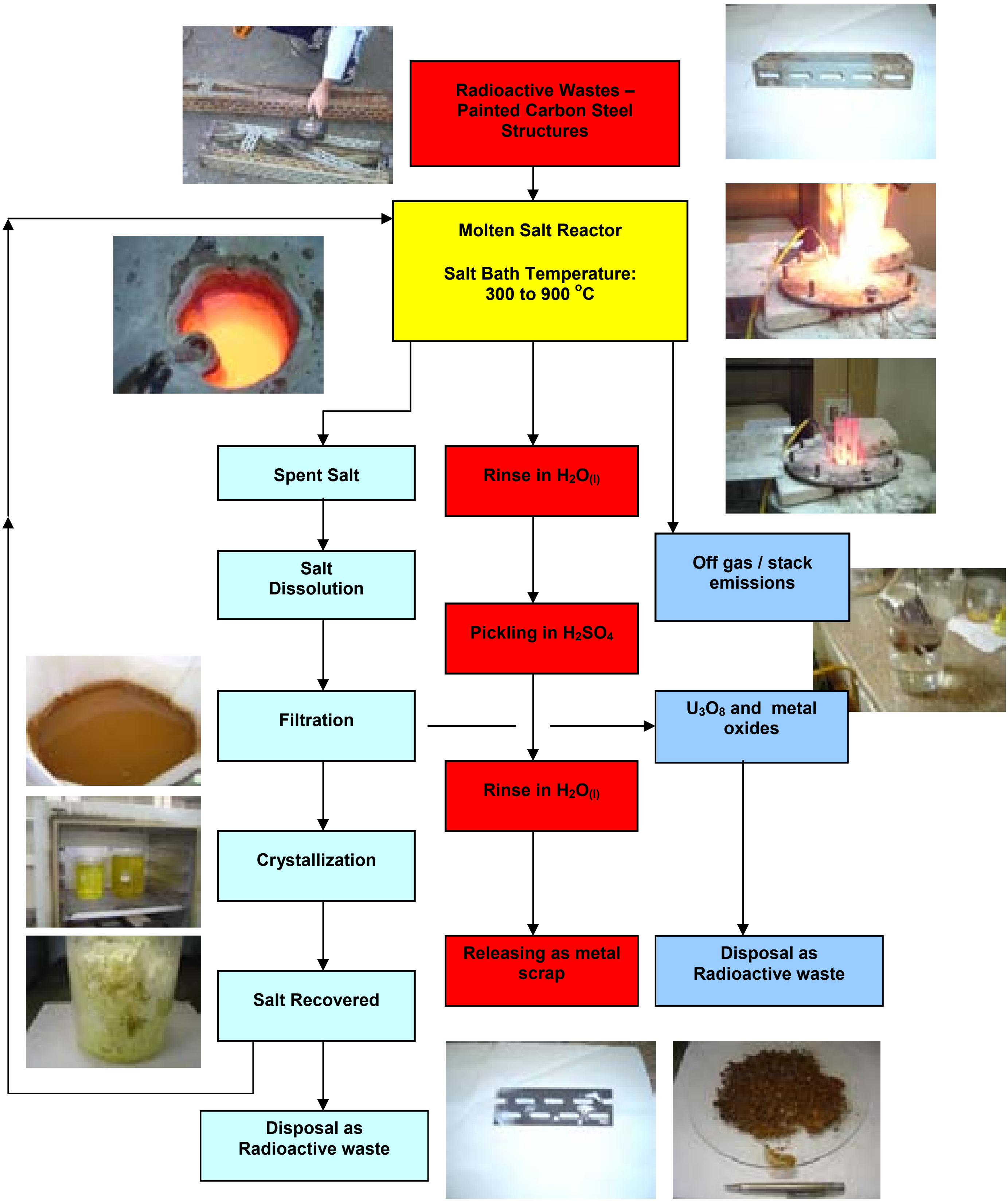

Fig.32. Flow sheet of the molten salt stripping treatment of radioactive wastes.

Different molten salt compositions have been tested to determine the best salt formulation since the temperature must be as low as possible. Sodium carbonate, sodium hydroxide, the eutectic mixture of sodium carbonate and sodium hydroxide and sodium hydroxide with addition of some oxidizing salts such as sodium nitrite were tested. The results obtained in terms of painting and contamination removal were excellent. Nevertheless, it is still possible to improve the salt mixture and optimize the salt temperature versus residence time. The exact mixture of salts can be tailored to achieve minimum operating temperatures and very short residence time. Dissolution of the salt, followed by filtration and crystallization of the salt aqueous solution by heating eliminates the liquid radioactive waste. The previously superficially contaminated parts can be fully released. The solid waste generated in the end of the process is only a small part of the initial volume. The solid waste retained in the filter presented 400 counts/s (maximum).

4. Interactions

The main interactions involving other members of the CRP have been:

- Russia/Sergey Mikheykin – visit of Sergey in IPEN in November 2007, with a lecture for IPEN's personnel; proposal of a visit to SIA RADOM for Paulo Lainetti in 2008 (IAEA Human Resources Development Project);
- Slovakia/Vladimir Daniska – visit of Paulo Lainetti, in November 30th, 2007 to the Radioactive Waste Treatment and Conditioning Complex in the Bohunice Site – Slovakia;
- Korea/Jin Ho Park – There is an umbrella agreement on nuclear energy between Brazil and Korea. Nevertheless, it is necessary to define the way to use the agreement to stablish an effective cooperation. The main points of interest defined in the preliminary contacts are dismantling techniques and waste management on the decommissioning of nuclear fuel cycle facilities. Both sides agreed to stablish cooperation between IPEN (Brazil) and KAERI (Korea) and to invite experts to their respective decommissioning sites in 2008;
- Czech Republic/Josef Podlaha - information supplied to Ing. Petr Kovarik about molten salt process and possibility of future cooperation in this field – Centre of Radioactive Waste Treatment / Nuclear Research Institute Rez and IPEN;
- Argentina/Silvio Fabbri - preliminary contacts about future visits in 2008 in Argentina and Brazil;
- Belgium/Jerome Dadoumont - preliminary contacts about future visit in SCK – CEN, Belgium.

Another important interaction was the Brazilian participation in the ENC 2007 – European Nuclear Conference, Brussels, September 2007, where it was presented ***Decommissioning of Nuclear Fuel Cycle Facilities in the IPEN-CNEN/SP*** (oral presentation), with the support of FAPESP (an Agency for R&D support from the São Paulo State Government/Brazil)**.**

5. Conclusion

The dismantling activities and the restoration of contaminated areas have been pursued consistently with the strategy of IPEN priorities and compatible with the available resources. For this work, the contribution of areas such as radiological protection, analytic support, wastes treatment, decontamination and mechanical/electric maintenance is indispensable. Particularly, the planning needs the involvement of different competences and skills. To the demand for these resources is added the lack of reliable information on the status of the facilities and the lack of a group specialized in the decommissioning area. The feasibility of UF_6 hydrolysis for removing material from small components is clearly possible. However, scale-up involving large equipment needs additional analysis to assure safe conditions and elimination of uncontrolled reaction excursions. Quickly expanding vapor and gases in confined equipment may be catastrophic.

Molten salt oxidation of the existent organic radioactive wastes was proven to be a reliable and feasible method of hazardous waste destruction. It will be necessary to promote some adjustments and optimize the process to operate inside glove-boxes. Analysis by CG/MS demonstrated that destruction efficiencies of 99.99986% are possible in the equipment developed. The only compounds present in the process off-gases are water and carbon dioxide.

The treatment of radioactive wastes in the form of painted structures made of perforated carbon steel, and presenting superficial contamination has been investigated. The superficial radioactive contamination is located mainly in the corroded regions and mixed with rust. This contamination is frequently difficult to remove due to the layers of paint applied on the contaminated area. The paint over the contamination impedes the action of the usual acid or alkaline pickling methods. The process selected for paint stripping was the immersion of pieces in molten salt mixtures at different temperatures and different residence times. Molten salt stripping uses simple and straightforward processing steps. The items to be stripped can be loaded into baskets or supported on hooks. The process allows rapid and complete paint removal with a minimum of handling. The method can be applied for parts with complex shapes. The internal superficial contamination of tubes, which are not reached by blasting or abrasive methods, can be successfully treated by immersion in molten salts.

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EXPERIENCE IN DECOMMISSIONING OF SMALL FACILITIES

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Abstract
This report summarizes the main results obtained by the Centre for Radiation Protection and Hygiene (CPHR) in the decommissioning of four small radioactive facilitates. These activities were carried out within the framework of the IAEA Coordinated Research Project (CRP) on Innovative and Adaptive Technologies in Decommissioning of Nuclear Facilities. It includes the decommissioning of a brachytherapy facility at the National Institute of Oncology and Radiobiology, the radiological characterization and decommissioning of a laboratory that used ^{14}C for research activities, the planning for decommissioning of an irradiator facility and the decontamination of a retention tank used for collection and control of liquid effluents generated at the Waste Processing Facility at the CPHR. Safety aspects have been taken into consideration in the four examples. Resource constraints have also been an important issue considered in defining the decommissioning strategy.

1. Introduction

Radioactive materials and radiation sources are widely used in Cuba in medicine, industry and research. There are, in total, nearly 200 radioactive facilities around the country. For different reasons some of these facilities had reached the end of their useful life and consequently they required decommissioning. The paper describes the actions taken for decommissioning of four radioactive facilities:

- A brachytherapy facility at the National Institute of Oncology and Radiobiology (INOR) that had been also used as temporary storage for disused sealed sources and which became contaminated with ^{137}Cs.
- A laboratory, in which non-sealed radioactive sources for research purposes were used.
- An irradiator facility at the National Centre for Animal Health (CENSA).
- A retention tank for collection and control of liquid effluents generated at the Waste Processing Facility at the CPHR.

Decommissioning activities were carried out within the framework of the IAEA Coordinated Research Project (CRP) on Innovative and Adaptive Technologies in Decommissioning of Nuclear Facilities. The purposes of decommissioning were different. The brachytherapy facility at the INOR and the laboratory needed to be released from regulatory control for use as non-nuclear facilities. The old use irradiator facility needed to be dismantled with the aim of installing a new irradiator. The retention tank at the Waste Processing Facility at the Centre for Radiation Protection and Hygiene (CPHR) was decontaminated in order to facilitate repair activities that were needed. A research contract was awarded to the CPHR, as part of the CRP. The contribution given by the IAEA was used in procuring equipment required for decontamination and dismantling activities.

2. Decommissioning of a brachytherapy facility at the INOR

At the beginning of the eighties in Cuba there was not in operation any centralized storage facility for radioactive waste. A remaining room belonging to a former brachytherapy facility at INOR was then used as storage facility for disused sealed sources arising from nuclear applications in medicine and industry. One or more ^{137}Cs sources stored in this area were leaking, causing radioactive contamination.

No regulations to address decommissioning were in place in the country at that time. This resulted in a lack of early decommissioning considerations and planning at the INOR. When the contamination was detected some attempts were carried out, but for different reasons, the requirements established by the regulatory authority, the National Centre for Nuclear Safety (CNSN) could not be achieved, and therefore the facility could not be released from regulatory control. The facility remained closed for more than 10 years because of the remaining contamination.

During the first dismantling and decontamination attempts, conducted in 1988, washing with water and detergent solutions was used for decontamination of walls and floors. This approach reduced significantly the radiation levels, but the use of water caused the spread of contamination to other areas, initially not contaminated, for instance the garden and the underground drainage pipes.

Other decontamination and dismantling activities were carried out in 1999 by the Radioactive Waste Management Group of the Centre for Radiation Protection and Hygiene (CPHR). Six ^{226}Ra sources were recovered. Radiation and contamination levels were significantly reduced by using chemical and physical methods for decontamination. Because of the high levels of contamination in the area and the strict requirements established by the Regulatory Authority for clearance (surface contamination-0.4 Bq/cm^2), the decommissioning of the facility would have been extremely expensive. This cost would be mainly due to the management of a large amount of generated very low level radioactive waste. For these reasons it was decided not to continue decontamination activities.

A new decommissioning strategy with more realistic dose criteria was then elaborated and presented to the Regulatory Authority for approval. The radiological criteria proposed for clearance in the decommissioning plan was: the annual dose received by members of the public should not exceed 0.3 mSv above the natural background, in the worst case scenario. The Regulatory Authority approved these criteria and the Institution received the authorization for decommissioning in 2003.

Significant remodeling works started in all the hospitals in 2004. The institution was therefore interested in concluding decommissioning of the former brachytherapy facility, as it was planned to reuse it for non-nuclear purposes. For this reason, it was necessary to start decontamination and dismantling activities in order to achieve the release of the facility from regulatory control. The Institution did not create proper funding for decommissioning. Consequently these activities were supported with limited financial resources provided by government authorities.

2.1. Description of the facility

Figure 1 shows a scheme of the contaminated rooms, located in the first floor in Section A of the Hospital [1]. The construction characteristics of these rooms were described in detail in the Decommissioning Plan. In general the floor in all the areas was laid with tiles. Below the tiles there was construction filling material, which was contaminated in some areas. Most of the rooms had brick walls covered with mortar and painted. Room 3 had a concrete wall, designed as shielding. Behind the wall there was a well that was used to store radioactive sources in the former brachytherapy facility. In Room 4 there was a working bench and two sinks. Room 6 was a toilet. The surface of the floor in all the areas was covered with plastic sheet. That was done when decontamination activities were stopped in 1999 to avoid the spread of remaining radioactive contamination [2].

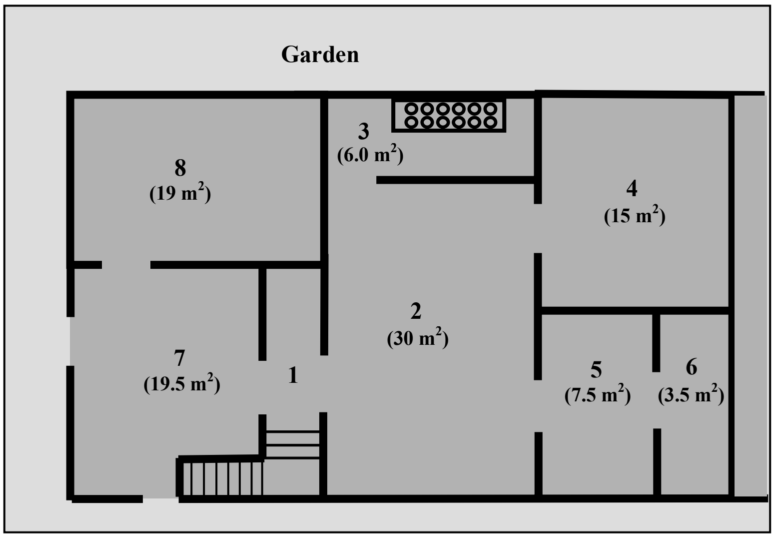

Fig. 1. Contaminated area at the brachytherapy facility in the Oncology Hospital.

2.2. *Initial radiological characterization of the facility*

For the initial radiological characterization, the dose rates at 5 cm from the surface of the floor were measured. Additionally, the dose rates at 50 cm and 1 m from the surface of the floor were also measured in the centre of the rooms. The initial radiological survey in the garden also included the spectrometric analysis of soil samples taken from different places. A smear test was used to evaluate removable contamination of walls and roofs. The maximum levels of specific activity reported in the soil of the garden were around 5 Bq/g. Table 1 shows the maximum dose rates measured at the surface (5 cm), and the dose rates at 50 cm and 1 m from the floor in the centre of each room.

Table 1. Maximum dose rates, in µSv/h, measured in the contaminated rooms and the garden

Area/ Room	Max. dose rate at 5 cm from the surface of the floor	Max. dose rate at 50 cm from the surface of the floor	Max. dose rate at 1 m from the surface of the floor
garden	6	2	1.5
1	13	7	4
2	196	30	8 – 40
3	46 – 220	50	30
4	7	2	1.5
5	28	2.5	2
6	1	0.6	0.6
7	22	3	2.0
8	5	0.6	0.6

The measurements of smear samples showed that there was no significant removable contamination on walls and roofs. Another conclusion from dose rate measurements was that the level of fixed contamination on walls must be low. The final evaluation of fixed contamination on walls and roofs was carried out after removing the contaminated floor and reducing the dose rate background.

The floor of the rooms 1, 4, 5, 6, 7 and 8 showed fixed contamination. The floor in room 2 had removable contamination in two zones where the tiles were very rough. The floor filling material in room 3 was contaminated, because some floor tiles had been removed during the first decontamination activities carried out in 1988. The well used to store brachytherapy sources had also removable contamination.

2.3. *Reference levels proposed to the Regulatory Authority for decommissioning of the facility*

The decommissioning plan contained the radiological criteria proposed to the Regulatory Authority for decommissioning of the facility: the annual dose received by members of the public should not exceed **0.3 mSv**, above the natural background, without the application of any restriction for the use of the facility. Following these criteria operational reference levels were derived:

- **Dose rate**: Dose rate at 10 cm from any surface (walls, floors tiles and roofs) should not exceed **0.1 μSv/h** above the natural background. Equation (1) was used to calculate the dose from a plane source. It was considered a two meter radius circular contaminated surface (^{137}Cs was considered as the reference contaminant) and dose rate 0.1 μSv/h at 10 cm. The dose rate at 100 cm and 50 cm would be 26.9 nSv/h and 47.3 nSv/h respectively. If a person stays in this room during a year, with occupancy factor 2/3, he/she would receive a maximum annual effective dose 0.22 mSv/year. This result was obtained as an example for a 2 m radius room. For smaller surfaces the dose rates would be lower.

$$E = \pi \cdot \Gamma \cdot A_s \cdot t \cdot \ln\left(1 + \frac{R^2}{a^2}\right) \qquad (1)$$

E ... effective dose received in the time (t) at the distance (a) from a plane surface with radium (R) and surface activity A_s.

Γ ... gamma constant for ^{137}Cs (8.6 x10-10 Sv./h.Bq^{-1}.cm^{2}), π… constant (3.14)

For contaminated soil (in the garden) and floor filling material (inside the rooms), the activity concentration was used to estimate the doses. This is explained below.

- **Activity concentration:** Specific activity in the soil should not exceed **1 Bq/g**. The planar dose from a contaminated material was calculated using equation (2). A material with density 1.6 g/cm^3 (typical soil) with activity concentration ρ=1Bq/g produces in the centre line of its area, with 4 m diameter, a dose rate of 48.1 nSv/h at 0.5 m from the surface and 34.5 nSv/h at 1 m. The dimensions of the area considered the real dimension of the garden. If a person stays over this surface for a year, with occupancy factor 2/3, he/she would receive a maximum annual effective dose 0.24 mSv/year. It was considered that the ^{137}Cs concentration was 1Bq/g in all the profile of the soil, but as the distribution coefficient (kd) is high, the activity concentration should decrease in depth. The occupancy factor, the dimensions of the contaminated area, as well as the depth of contaminated soil were estimated, taking into account conservative assumptions. It is expected that the annual doses would be lower than the calculated.

$$E = 2 \cdot \pi \cdot C_A \cdot \rho \cdot \Gamma \cdot \int_0^R \int_0^H \frac{t \cdot r}{(x+a)^2 + r^2} \cdot \exp\left(-\mu \cdot x \cdot \left(\frac{\sqrt{(x+a)^2 + r^2}}{x+a}\right)\right) \cdot dx \cdot dr \qquad (2)$$

The commitment effective dose from a hypothetical consumption of food produced in the area of the garden, contaminated with ^{137}Cs (1Bq/g), was evaluated for the following scenarios: ingestion from the consumption of milk and meat, considering the pasture in the area of the garden and ingestion of vegetables directly produced in the garden. The doses obtained from the assessment were trivial because of the low concentration of the radionuclide (1Bq.g-1), as well as the small extension of the contaminated area.

2.4. *Strategy proposed for decommissioning of the facility*

The strategy for the dismantling and decontamination (D&D) was described in the Decommissioning Plan. Radioactive sources had been removed from the facility in previous D&D activities [1] and most of existing contamination was fixed, therefore dismantling activities were those mainly considered in developing the strategy for decommissioning. The dismantling techniques were very simple, consistent with the type of facility and contaminated materials, as well as the low contamination levels.

Dismantling work consisted in removing: the soil from the garden with activity concentration above 1 Bq/g; contaminated tiles, construction filling materials and parts of the surfaces of the walls with contamination levels above the reference dose rate (0.1 µSv/h above the natural background at 10 cm from any surface), pipes and other contaminated objects.

The task consisted in a cycle of repeated activities: evaluation – decontamination – dismantling – evaluation, etc; in which the requirements for waste management, radiological control to the public and workers, control of discharges of radioactive materials to the environment as well as emergency planning were taken into consideration. In order to reduce the dose rates, D&D activities started in the most contaminated zones and rooms. This was important for radiological protection and also to facilitate the evaluation of contamination levels in less contaminated areas. The spread of contamination to other areas was avoided. The important principle of minimization of radioactive waste was also considered. All radioactive wastes generated from decommissioning activities were transported to the Waste Management Facilities, authorized by the Regulatory Authority.

The doses to be received by the operators were previously estimated. Radiological surveillance of potentially exposed individuals was maintained, as well as the monitoring of working areas, in order to verify the compliance with the dose constraints established in the Decommissioning Plan. The monitoring activities included: dose rate levels, surface contamination, activity concentration in aerosols and individual doses from personal dosimetry.

The strategy for decommissioning with the new radiological criteria was presented to the regulatory authority (CNSN) for evaluation. The CNSN approved these criteria and the INOR received the authorization through a license for decommissioning.

2.5. Description of D&D activities

The objective of D&D activities was the release of the facility from regulatory control. For that it was necessary that radiation and contamination levels be below the approved reference levels, allowing the further use of the facility. The operations for decommissioning of the facility were accomplished according to the Decommissioning Plan and the Safety Manual.

Simple dismantling techniques were used, consisting of removing and segregating contaminated tiles, floor filling materials, soil, etc. where dose rate, contamination levels or activity concentration were above the approved clearance levels (Fig. 2). It was also necessary to cut parts of the walls in room 3 and sink and working benches in room 4.

Fig. 2. Dismantling operations inside the rooms.

An important factor was the minimization of radioactive waste. All contaminated materials (tiles, scraps, filling material, etc.) removed from the area were placed in plastic bags. Each bag was monitored and segregated according to the radiation levels at the surface. Materials with lower contamination were segregated from the rest in order to evaluate in the future, the possibility of conditional clearance. All radioactive waste was transported to the Centralized Waste Storage Facility.

2.6. Necessity for changing the initial strategy

Although considerable amount of contaminated material had been removed from the area, the reference levels established for decommissioning could not be achieved. For example, the room 3 was the most contaminated. All the floor tiles, the filling material and a concrete layer of 10 cm approximately had been removed. Below the concrete there was soil. A sample of this material was taken, measured and showed to be contaminated. A study of the activity concentration in the profile of the soil was carried out. An area of 30 cm x 30 cm was selected in the centre of the room, where the concrete had no crack that could have allowed (in the past) the direct entrance of contaminated water. Samples of concrete and soil up to 20 cm depth approximately were taken. The samples were analyzed in the gamma spectrometric system. The results are shown in Table 2.

Table 2. Results of the analysis of concrete and soil samples in room 3

Code of the sample	Information about the samples	Specific activity of ^{137}Cs, Bq/g	Remarks
H1	Concrete, from surface up to ~ 5 cm depth	20.5	Above the clearance level
H2	Concrete, from ~ 5 cm depth up to the soil	6.9	Above the clearance level
T1	Soil, 5 cm layer, below the concrete	78.2	Above the clearance level
T2	Soil, 5 cm layer, below T-1	14.2	Above the clearance level
T3	Soil, 5 cm layer, below T-2	4.2	Above the clearance level
T4	Soil, 5 cm layer, below T-3	0.66	Below the clearance level

According to these results, and for achieving the reference level (1 Bq/g, in terms of activity concentration), it would be necessary to remove all the concrete from the floor of room 3 and a layer of soil of about 15 cm depth. That would generate a considerable amount of radioactive waste.

A similar situation occurred below the doorframes in most of the rooms as well as in some areas of the garden. The picture in the Figure 3 shows, as example, the dose rate levels remained in a contaminated zone in room 7. A considerable amount of contaminated material had been removed: the floor tiles and the construction filling material up to 20-40 cm depth. The radiation levels were significantly reduced but the reference level for decommissioning in terms of activity concentration was not achieved.

Fig. 3. Dose rate levels (μSv/h) remained in a contaminated area of room 7.

As the dose rate levels were not significant (7 µSv/h was the maximum dose rate at the surface of the holes), and continuing removing contaminated soil would generate a considerable amount of very low level radioactive waste, the strategy for decommissioning was then change to entombment [3]. That strategy was based on the assumption that some construction works were needed for the release of the facility from regulatory control. The "holes" must be filled with soil or other materials, which at the same time would serve as shielding. The depth of the holes was calculated in order to guarantee that after filling them with new material the reference level in terms of dose rate would be achieved.

Special consideration was given to the underground components, such as drainage pipes. As appropriate equipment for characterization of underground pipes was not available, the activity was estimated using the model for a lineal source. The procedure was described in detail in reference [4]. No drawings or technical details about the underground pipes were available at the facility. It was assumed that pipes were located in rooms 4, 7 and 8 (Fig. 1). After contaminated tiles and soil (filling material) were removed from these rooms, the dose rate at 10 cm from the surface of the floor was measured. Having reached the criteria for clearance (0.1 µSv/h), it was assumed that the pipes were not contaminated or the contamination levels were very low. For this reason it was decided to leave the pipes in the facility.

2.7. Description of the final radiological situation in the facility

Once D&D activities were concluded in all the areas, a final radiological survey was carried out. It included dose rate measurements at the surface of floors, walls and roofs. The reference level in terms of dose rate was achieved in almost all the areas, except around the doorframes, where the dose rates at floor level was around 1.0µSv/h.

Because of the dilution of ^{137}Cs in the water used during previous decontamination activities (1988) and its penetration through the fissures existing in the floor (for example, below the doorframes), it was not reasonable to achieve the reference levels (0.1µSv/h at 10 cm from the surface) in these zones only by removing contaminated materials. The long period that decommissioning of the facility was deferred had also a negative influence. Nevertheless, in the final radiological evaluation it was considered the final situation in the facility, after the necessary construction works.

Taking into account that the half value layers (HVL) and ten value layers (TVL) of ordinary concrete (2.35 g/cm^3) are 4.8 cm and 15.7 cm respectively for the energy of ^{137}Cs, it was possible to assume that if the holes are filled with concrete (more than 20 cm in all the zones), the dose rate levels would be reduced more than 10 times. Consequently, the dose rates at the surface of the floor would be less than 0.1µSv/h. Considering the dimensions of the rooms, the dose rates increments (above the natural background) at 50 cm and 100 cm from the surface were estimated. The dose rates would be further reduced after the floor of the room is laid with tiles. The results of dose rate estimations are summarized the Table 3. The dose rates measured before and after dismantling are also showed in Table 3.

The annual dose from the reuse of the facility was also estimated. Two situations were considered for evaluation: residential condition (the exposed person lives in the room), for which the occupancy factor is 2/3; and working condition, the exposed person is inside the room 8 hours per day, during 5 days a week and 50 weeks a year. As expected, the estimated annual effective dose in all the rooms and in the garden was below the radiological criteria approved for decommissioning: 0.3 mSv/year [2].

Table 3. Dose rates measured at the centre of the rooms before and after D&D and estimated after final construction works

Room	Distance from the floor	Before dismantling, µSv/h	After dismantling, µSv/h	After filling the holes, nSv/h	After laying the floor with tiles, nSv/h
1	50 cm	7.0	0.33	35.7	18.0
	1 m	4.0	0.32	15.4	8.0
2	50 cm	30.0	0.25	61.2	30.0
	1 m	20.0	0.20	44.4	21.0
3	50 cm	50.3	0.39	51.9	25.0
	1 m	30.0	0.32	32.2	15.0
4	50 cm	2.0	0.08	-	15.0
	1 m	1.5	0.07	-	10.0
5	50 cm	2.5	0.07	-	10.0
	1 m	2.0	0.06	-	5.0
6	50 cm	0.6	0.1	-	30.0
	1 m	0.6	0.09	-	17.0
7	50 cm	3.0	0.2	63.0	32.0
	1 m	2.0	0.18	43.7	22.0
8	50 cm	0.6	0.07	-	10.0
	1 m	0.6	0.06	-	5.0
Garden	50 cm	2.0	0.2	27.0	-
	1 m	1.5	0.18	25.0	-

2.8. Position of the Regulatory Authority

Once the dismantling and decontamination activities were finished a radiological survey was performed in the facility and the final report was presented to the regulatory authority. The new strategy adopted for decommissioning was described and presented for approval.

The regulatory authority evaluated the proposal and carried out an inspection to the facility. It was considered that dismantling and decontamination activities could be stopped, taking into consideration that subsequent activities would not achieve significant reductions in the radiation and contamination levels. The following (revised) requirements for decommissioning were established by the Regulatory Authority:

— The dose rate at 10 cm from any surface (floors and walls) should not exceed 0.1 µSv/h above the natural background. Necessary shielding should be guaranteed where this level has not been reached.
— Contaminated materials should be isolated from human contact. Regarding the underground pipes, it was required that no maintenance or repairing activities were to be performed in the contaminated areas (mainly around the traps). Thus the drainage in the contaminated area should be closed and new ones must be constructed outside this space. The existing drainage should not be used any more.

2.9. Final construction works and decommissioning

Reconstruction activities were needed in the facility in order to comply with the requirements established by the regulatory body to achieve the final release of the facility from regulatory control. The CPHR presented a report to the regulatory body and to the contractors explaining in detail the construction work. The report also contained a radiological evaluation for these operations. In general the operations included:

— To fill with concrete (2.35 g/cm^3) the holes opened in the floors up to approximately 10 cm below the floor level.

— To fill with construction filling materials up to the top of the holes and level all the floors of the rooms.
— To tile the floor and level the walls.
— To not install sewer nor electrical systems below the floors.

All operations were supervised by a radiation protection specialist. After concluding the reconstruction activities, dose rates were measured throughout the facility. They were below the reference levels established by the regulators.

The facility was released from regulatory control. A new department for healing of patients after different cancer surgeries was established in these recovered rooms (Fig. 4).

Fig. 4. Reuse of the facility after decommissioning.

2.10. Management of radioactive waste generated from decommissioning

The following amounts of radioactive waste were generated during decommissioning activities at the Oncology Hospital:

— Non-compressible solid radioactive waste: 18 m^3
— Compressible solid radioactive waste: 1 m^3
— Non radioactive waste: 5 m^3

The strategy for the management of radioactive waste has also been influenced by limited financial resources. Compressible radioactive waste did not represent any problem, as the amount was very low and it could still be reduced. However, the volume of non-compressible solids represented approximately 20% of the operational capacity of the centralized storage facility. Moreover, if these wastes were conditioned, the volume would be increased by at least twice.

As most radioactive waste generated during decommissioning were very low level waste, conditional clearance and release have been considered as an appropriate strategy for their management. A detailed characterization of generated radioactive waste was necessary for defining the possibility of clearance. During D&D activities, non-compressible solid wastes were collected in plastic bags and stored in 90 standard 200-liter drums. The total activity in each drum was estimated from the maximum dose rates measured at a certain distance from the drums, according to the methodology described in [5] for extended sources and cylindrical geometry. The total activities were between 15 and 500 MBq. Specific activities were also estimated: they varied between 40 and 1600 Bq/g [3].

The clearance of radioactive waste with very low activity and the release of cleared materials to a conventional landfill were considered the most appropriate way for the management of this type of waste. The scenario for conditional release considering the human intrusion as an important factor has been carefully selected. As the wastes contain soil and debris, it was recommended to release them or deposition on a specific route between the existing trenches. The releases materials would be transported in a truck to the landfill, then the materials are to be released or discharged from the truck to the roadway and then it would be necessary to homogenously distribute them along the road. This zone was less frequented by intruders. In order to assess the radiological impact, the following scenarios were considered:

(1) The release of radioactive materials on the road between the trenches
(2) The deposition of the material along the road
(3) Human intrusion in the landfill

The radiological impact assessment for each scenario, as explained in reference [3], was carried out. The basic for calculation was an annual dose limit of 10 μSv for the public in the considered scenarios. It was obtained that 49 drums contained concrete debris and soil with activity concentration less than 192 Bq/g could be considered for conditional clearance and release. This option will be presented to the Regulatory Authority for evaluation.

2.11. Lessons Learned and concluding remarks

- Decommissioning considerations and planning are very important aspects that should be considered from the very beginning, i.e. from the design, construction and commissioning of the facility. It is also essential to maintain drawings and appropriate records about the construction and operation of the facility.
- Wash cleaning using considerable amount of water or any solution for decontamination, should be carefully evaluated in advance. This method is not always efficient because even when it could be effective to reduce gross contamination levels, the spread of contamination makes it difficult to achieve the approved decommissioning levels, mainly when soluble compounds, like cesium salts, are involved.
- The conditional clearance of very low level radioactive waste is a valuable solution that should be considered

3. Decommissioning of a radiochemical laboratory at the International Centre for Neurological Restoration

The International Centre for Neurological Restoration (CIREN) used ^{14}C non-sealed sources for basic radiochemical research. This practice was authorized by the Regulatory Authority through a registration granted in 2003 and valid for 4 years. The institution decided to use an alternative non-radioactive technique for the same research purposes. For this reason, the practice was terminated and the institution requested the release of the radiochemical laboratory from regulatory control. For the decommissioning activities the CIREN contracted the CPHR services, which began with a radiological survey and characterization of the facility.

Regarding the operational record keeping, it was fortunate that the historical records were generally very good. A thorough review of the available documentation (authorization, source inventories, inspection reports, radioactive waste collection reports) revealed that the main radionuclide used in the last years was ^{14}C, but between 1993 and 1996, other radionuclides, such as ^{3}H, ^{51}Cr, ^{125}I and ^{32}P were used. The material containing very short lived radionuclides had decayed to negligible levels. Tritium material was of very low activity and most of the material was previously collected as radioactive wastes. Therefore, from the decommissioning perspective, only ^{14}C was considered as radiological inventory. Because of the radiological characteristics of this radionuclide, the reference levels for decommissioning were considered only in terms of surface contamination (fix and removable).

The clearance levels for materials containing radioactivity are established in the regulation [6] in terms of activity concentration. For ^{14}C the clearance level is 30 Bq/g. For the radiological survey and decommissioning purposes it was necessary to derive the surface contamination limits, which enables qualifying the exposure risk due to removable and/or fixed surface contamination. It was also necessary to define how to measure the derived surface contamination figures.

3.1. Derived reference levels for radiological characterization and decommissioning

Reference [7] contains the derived limits in terms of surface contamination recommended for low and intermediate activity laboratories. These limits were estimated considering an annual dose for occupational exposed workers of 20 mSv and an annual occupational exposure of 2000 hours. The reference levels for ^{14}C recommended in this reference are:

- Removable surface contamination: 400 Bq/ cm^2
- Fixed contamination: 40000 Bq/ cm^2

Taking into consideration that the International Centre for Neurological Restoration will request the unrestricted release of the facility from regulatory control, the following reference levels were recommended based on a small fraction of contamination limits recommended in [7]:

- Removable surface contamination: 4 Bq/cm^2. This value is 100 times lower than the value recommended for occupational exposed workers.
- Fixed contamination: 2000 Bq/cm^2. This value is 20 times lower than the value recommended for occupational exposed workers.

3.2. Radiological characterization and decommissioning

A wipe test was used for assessing removable contamination in the most probable contaminated areas: work benches, glove box, refrigerator, etc. (Fig. 5). The samples were measured by liquid scintillation counting.

Fig. 5. Radiological characterization activities in the laboratory.

Because of the difficulties to directly measure weak beta emitters such as ^{14}C, it was necessary to agree a practical approach with the regulator. The existing surface contamination monitor (Mini-Instrument LTD, model 1500, with probe DP2R/4) was not calibrated to measure ^{14}C. The equipment was then calibrated using a reference surface sample with the derived contamination limit (2 kBq/cm^2), prepared for simulating the surface contamination. Once the response limit for the used specific monitor was defined (15cps), the surface contamination survey was carried out. All the surfaces (work benches, floors, walls, etc) were measured.

By using both techniques, direct monitoring and wipe testing (with liquid scintillation counting), it was possible to accomplish the monitoring tasks. We considered that 10% of the non-fixed contamination is removed with the wipe, for all the surfaces. The results of characterization in terms of removable and fixed surface contamination were below the established reference levels. Therefore no decontamination activities were needed in the facility.

A report with radiological characterization of the facility was presented to the Regulatory Authority, who reviewed the report and approved the unrestricted release of the facility from regulatory control.

3.3. Lessons learned and concluding remarks

- Because of the difficulties to directly measure weak beta emitters such as ^{14}C, and the lack of appropriate equipment, it was necessary to agree on a practical approach with the regulator, which included the monitoring and decontamination methodologies. Practical solutions were applied to solve the lack of specific equipments and tools.
- The decommissioning of the research laboratory at CIREN demonstrated the value of maintaining appropriate records about the operation of the facility.

4. Decommissioning of an irradiator facility used for research purposes in Cuba

The National Centre for Animal Health (CENSA) has an irradiator facility, model Gammacell 500 from MDS Nordion, which was used for sterilizing medical and pharmaceutical devices and supplies. The facility was commissioned in 1989. Although the facility was in use, it had to be shut down, as the safety mechanisms might become ineffective. A new irradiator facility was to be procured and installed in the same place.

The decision for shut down was taken. According to Cuban regulations, the institution CENSA had to apply for a decommissioning license. For that, the following documents had to be presented to the Regulatory Authority: plan for shut down, decommissioning plan and safety manual. The institution requested these services from the Centre for Radiation Protection and Hygiene (CPHR). The documents noted are currently under preparation.

4.1. Facility description

The irradiator facility Gammacell 500 has twelve ^{60}Co radioactive sources, model C-198. The total activity was 1069 TBq (28900 Ci) in December 1994, after the sources were replaced by the manufacturer. The total activity in 2007 was around 200 TBq.

The irradiator has a sample chamber or cylinder (Fig. 6). The sample trays rotate around the sources when they are loaded to this chamber (while in the irradiate position), providing dose uniformity. The cylinder (sample chamber) measures 1524 mm wide by 1380 mm in length by 1695 mm in height and the maximum weight is about 16 tons.

Fig. 6. Irradiator facility Gammacell 500, sample chamber.

Fig. 7. Irradiator facility Gammacell 500, source storage container (the smaller cylinder).

The source storage container is located behind the sample chamber (Fig. 7). It is a lead cylindrical container which measures 1016 mm wide by 1420 mm in length by 1003 mm in height and the maximum weight is about 6.3 tons.

A remote mechanism is used to move the sources from the storage container to the sample chamber for the irradiation position and, vice versa, when irradiation is finished. After several years of operation, the remote mechanism began deteriorating, which required more frequent maintenance. The leak test carried out last year has proven that the sources are not leaking.

4.2. Decommissioning strategy

The strategy for decommissioning consists in removing the high activity ^{60}Co sources from the irradiator facility. First the sources have to be fixed in the storage container and then this container should be removed from the unit and placed in an overpack. The package should be transported to the Centralized Storage Facility for Radioactive Waste. A special authorization of the Regulatory Body is needed for transport. The possibility of returning the sources to the manufacturer should be investigated.

After removing the sources, a detailed radiological survey will be carried out in the facility and the rest of the unit will be dismantled. No radioactive contamination is expected.

In 2008, the irradiator facility is expected to be decommissioned as a new facility will be installed in the same building. The decommissioning plan is under development at present.

5. Radiological characterization and decontamination of the retention tank used for collection and control of liquid effluents at the Waste Processing Facility

The current Radioactive Waste Processing Facility (WPF) was constructed in a building formerly used by the Centre for Distribution of Radioisotopes (ENSUFARMA), where radioactive materials were handled. The facility had a system for collection and control of generated liquid effluents, but it did not fully comply with the functions for which it had been designed. The storage tank for collection of liquid effluent (Fig. 8) did not fulfill the technological requirements for a radioactive facility Category I, as the Waste Processing Facility is considered.

Fig. 8. Retention tank for collection of liquid effluents at the WPF.

Liquid effluents have been stored in this tank from the beginning of the nineties, when the facility did not belong to the Centre for Radiation Protection and Hygiene. No records or information were available regarding the liquids stored at that time. Later on an additional problem was detected: the tank did not retain the liquids due to some fissures at the bottom surface.

During routine operations at the WPF (before the fissures in the tank were detected), radioactive liquid effluents were generated and collected in the retention tank. Consequently the tank was radioactively contaminated. Decontamination was then necessary before repairing. As there was no information available regarding the design and construction of the retention tank, or the liquids stored in the past, an additional characterization and investigation was needed for an adequate planning of decontamination and repairing activities.

5.1. Site characteristics

5.1.1. General characteristics of the canalization system

The canalization system has four underground compartments (Fig. 9) located beside the main building of the Waste Processing Facility. Liquid effluents generated at the WPF come into these compartments through underground pipes.

The retention tank (compartment no.2) was the one that represented the most interest for radiological characterization. The retention tank had to be repaired in order to eliminate the fissures existing at its bottom surface. Before repairing it, it needed to be decontaminated.

Compartments 1 and 3 contain the valves by which the effluent movement is controlled and directed either to the retention tank (2) or to the external non-radioactive outlet (4). The connection pipes are slightly contaminated, but the contamination levels are negligible and they do not represent any irradiation or contamination risk for the public and the environment.

The compartment 4 is the connection to the outside. It presented a non-fixed contamination that had been previously removed.

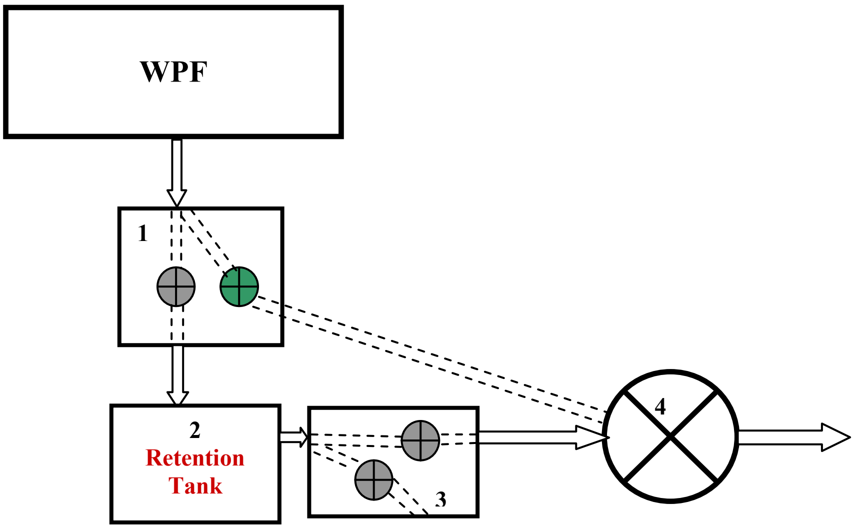

Fig. 9. Scheme of the canalization system at the WPF.

5.1.2. Construction characteristics of the retention tank

As already mentioned, no information related to the design and construction aspects of the retention tank was available. During the characterization, it was identified that the tank had concrete walls and floor, with a smooth painted cement mortar finishing.

The dimensions of the retention tank are: 170 cm long by 170 cm wide by 192 cm deep. It had a heavy metallic cover (lid) that could be removed during operations inside the tank (characterization, decontamination and repairing) and then returned to the initial position.

5.1.3. Operational history

No records were available about the operational history of the canalization system since it was commissioned. An unexpected volume reduction of stored liquids in the retention tank was detected at the end of 2006. This led to suspicion about possible leakage through the bottom of the tank. Since this deficiency was detected, the operations in the retention tank were temporarily stopped. Consequently, the activities in the Waste Processing Facility were drastically reduced. Just few relevant operations were authorized, such as collection and storage of radioactive waste from external generators and the characterization, decontamination and repair of the retention tank. A temporary "alternative canalization system", with a plastic container was then used in the laboratory for collection and control of generated effluents (Fig. 10).

Fig. 10. Temporary system for collection of possibly contaminated effluents at the WPF.

5.2. Initial radiological characterization. General issues considered

5.2.1. Objective of characterization

Characterization is an initial step in (D&D) process and requires a logical approach in order to obtain the data necessary for planning and implementing a D&D programme [8]. The radiological characterization included measurements of liquid and sludge samples taken from the tank, direct monitoring of dose rate and surface contamination, as well as in-situ spectrometric measurements inside the tank. These operations, as well as the associated records were documented in procedures and reports.

5.2.2. Radionuclide inventory

In the most recent years, radioactive wastes contaminated with very short lived radionuclides (T1/2 < 100 d), and other radioactive waste and disused sealed sources contained shorter lived (^{137}Cs) and longer lived (^{226}Ra, ^{241}Am) radionuclides had been handled at the Waste Processing Facility. During these operations liquid effluents generated were collected in the retention tank.

In August 2005, during the routine sampling and characterization of liquid effluents stored in the retention tank prior to discharge, it was realized that these liquids were contaminated with ^{137}Cs and ^{226}Ra. This brought the suspicion that the retention tank could be contaminated with these two radionuclides.

5.2.3. Criteria proposed for decontamination

Clearance levels in terms of surface contamination on building surfaces and process equipment etc. have not been established yet in Cuban regulations. After decontamination and repair, the retention tank would be used for the same purpose as it had been used until now. Therefore, since it was located inside the control area of the Waste Processing Facility, the following criteria were proposed as end points for the decontamination operations:

- To remove non-fixed contamination from the floor and walls of the retention tank.
- To remove non-fixed contamination from the outlet pipe and in the inlet pipe as far as possible.
- To remove fixed contamination on the walls and floor of the retention tank until the dose rate on the surface level was 1 µSv/h (this is the dose rate limit outside the building of the WPF established in the License).

5.2.4. Monitoring techniques

The samples taken during characterization were measured at the Laboratory of Environmental Radiological Surveillance of the CPHR, using a high resolution gamma spectrometric system with HpGe detector.

A portable gamma spectrometer with a NaI detector was used for in-situ measurements (Fig. 11). Two spectra were taken in the centre of the retention tank, from the surface (1.9 m from the bottom) and inside the tank (20 cm from the bottom).

Fig. 11. Spectrometric measurements in the retention tank to identify the radionuclides.

Dose rates were measured using a RADOS monitor, model RDS-110. This equipment was metrologically verified at the Secondary Standard Dosimetry Laboratory (SSDL) of the CPHR. Surface contamination measurements were carried out using a RADOS monitor, model MicroCont II. The calibration of this monitor for these specific operations is described below, in Section 5.2.5.

5.2.5. Calibration of the surface contamination monitor

For the calibration of the surface contamination monitor MicroCont II, ^{137}Cs and ^{57}Co certified standard sources were used. The measurements for calibration were performed at 1 cm from the surface of the plane sources and with the plastic protection placed on the detector. It was known that the retention tank was contaminated with ^{137}Cs and ^{226}Ra. Since ^{226}Ra calibration sources were not available, a ^{57}Co plane standard source was used for calibration. This radionuclide has emission energy for gamma radiation very close to the emission energy of ^{226}Ra.

For the ^{137}Cs standard source, the calibration coefficient obtained was 0.909 Bq.cm^{-2}/cps and for the ^{57}Co standard source it was 0.477 Bq.cm^{-2}/cps.

5.3. Description of radiological characterization operations and obtained results

5.3.1. Preparation of the control area

An area for the control of personnel and materials taken from the tank was prepared in front of the retention tank. This zone was covered with a water-resistant plastic sheet (Fig. 12). The metallic cover of the tank was removed and placed on the street in an inverted position.

Fig. 12. Control area prepared for characterization and D&D operations in the retention tank.

5.3.2. Measurements of the natural background

The natural radioactive background outside the Waste Processing Facility is not influenced by the waste stored inside. The natural background measured with the equipment used is:

- Dose rate (RADOS): 0.1 µSv/h
- Contamination (MicroCont II): 18 - 20 cps

5.3.3. Collection of liquid effluents and sludge from the retention tank

The retention tank contained liquid effluents and sludge that were collected using the vacuum cleaner and poured into a plastic container (Fig. 13). Samples of these effluents were taken for spectrometric analysis. The dose rate at the surface of the plastic container was 8 µSv/h. The effluents were stored as radioactive waste.

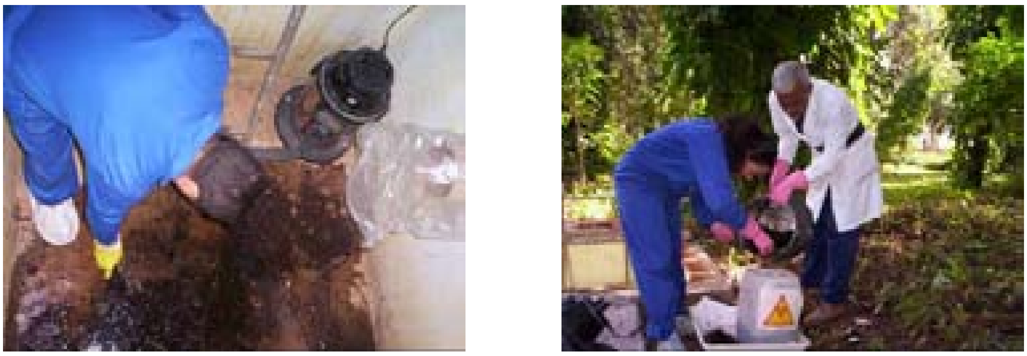

Fig. 13. Collection and storage of radioactive effluents from the retention tank.

5.3.4. Direct dose rate and in-situ spectrometric measurements

Dose rates were measured at approximately 20 cm from the bottom of the tank, the maximum values obtained were:

- Before removing the effluents: 8 µSv/h in the centre of the tank and 6 µSv/h in one of the corners.
- After removing the effluents: 5 µSv/h in the centre of the tank and 4 µSv/h in the corner.

After removing contaminated effluents, the dose rate in the upper surface of the retention tank was 0.24 µSv/h (in the centre). In-situ measurements using the portable gamma spectrometric system with NaI detector confirmed that the radionuclides presented were ^{137}Cs and ^{226}Ra.

5.3.5. *Surface contamination monitoring*

The wall surfaces and the floor of the retention tank were divided into 16 rectangular grids (Fig. 14) for the radiological survey. The area of each grid ranged from 0.15 m^2 to 0.2 m^2.

Fig. 14. Preparing the walls and floor of the retention tank for radiological survey.

In order to measure the surface contamination, the area with highest contamination levels inside each grid was first identified. Then the collimator was placed in this area and the measurement was repeated (Fig. 15). The detector was used with the plastic protection. The measurements were done at 1 cm from the surfaces (the same conditions as used for calibration of the monitor).

Fig. 15. Surface contamination monitoring inside the retention tank.

The results of the surface contamination monitoring (in cps) were recorded. The previously obtained calibration coefficients (0.909 $Bq.cm^{-2}$/cps for ^{137}Cs and 0.477 $Bq.cm^{-2}$/cps for ^{226}Ra) were used to estimate the surface activity (Bq/cm^2) from the equipment measurement units (cps). The ratio between the activity for each radionuclide with regard to the total activity was also considered for this estimation. From the spectrometric analysis of concrete samples taken from the bottom of the retention tank it was known that the activity concentration of ^{137}Cs was 30.4 kBq/kg and for ^{226}Ra, 21 kBq/kg. Therefore it was considered that:

Activity of ^{137}Cs / Total activity = 30.4 kBq/kg / (30.4 + 21) kBq/kg = 0.59

The 59 % of the total activity corresponds to ^{137}Cs, and the remaining 41% belongs to ^{226}Ra. Following this procedure it was obtained that the surface activity concentration for ^{137}Cs varies from 15 to 800 Bq/cm^2 and for ^{226}Ra from 5 to 300 Bq/cm^2.

5.4. Description of dismantling and decontamination operations

5.4.1. *Strategy adopted for D&D activities according to initial radiological characterization*

Dismantling and decontamination operations started from the floor, as it was the most contaminated area. The concrete layer was removed using an electric hammer. The radioactive contamination in the construction filling material (soil below the concrete layer) was assessed.

A hand held concrete grinder was used for decontamination of walls. The electric hammer was used in most contaminated areas. The wall surfaces were cut away in thin layers until all contamination was removed. A vacuum cleaner was used to avoid the dispersion of radioactive powder.

5.4.2. Description of D&D operations

Dismantling and decontamination operations started from the most contaminated grid locations of the floor: P14, P13, P9, P10, P5 and P6. The concrete layer in each grid was removed (the thickness varied from 5 to 6 cm). The 5 cm soil layer placed below the concrete was also removed (Fig. 16).

Fig. 16. Dismantling operations in the bottom of the retention tank.

After removing the entire floor, the more contaminated areas were located in the inlet and outlet pipes. They were covered with lead sheets and the surface contamination was measured in the floor and walls. High contamination was found in the lower grills of the walls, as these parts were in contact with contaminated effluents. They were removed using the electric hammer (Fig. 17).

Fig. 17. Dismantling operations in the walls.

The next step was to decontaminate the pipes. First, the radiation and contamination levels were measured (Fig. 18).

Fig. 18. Measuring radiation and contamination levels in the pipes.

The vacuum cleaner was used to remove contaminated powder. Some tools, such as a scrubbing brush and decontaminating solutions (detergent solution and commercial Radiacwash) were used for decontamination. Smear samples were used to evaluate the removable contamination (Fig. 19).

Fig. 19. Decontaminating the pipes.

The outlet pipe was completely decontaminated. The removable contamination in the inlet pipe was eliminated only near the entrance to the retention tank. Contamination levels were very low, so further dismantling operations in the inlet pipe were not justified. It was considered that the retention tank would be still used for collection and control of liquid effluents generated at the WPF. Therefore the possibly contaminated material removed from the inlet pipe would go to the retention tank for evaluation.

5.4.3. Final radiological situation in the retention tank

A final radiological monitoring was carried out after finishing the decontamination of the retention tank. The fixed surface contamination for ^{226}Ra varied from 6 to 16 Bq/cm^2 and for ^{137}Cs from 16 to 43 Bq/cm^2. The measurements of smear samples showed no removable contamination. In the inlet pipe the surface contamination monitor showed values between 330 and 1000 cps.

Radiation levels at the surface of the retention tank corresponded to the natural radioactive background.

5.5. Radiation safety measures during characterization and D&D operations

The required means for individual protection (protective clothes, gloves, overshoes, caps and mask) were available for characterization and decontamination operations in the retention tank. Before starting the operations, a control area was prepared in front of the retention tank, as described in 4.1.

In order to avoid the dispersion of contaminated powder, a vacuum cleaner was used during dismantling activities (Fig. 20). The whole body counter was used for measuring the internal contamination of the operators before and after D&D.

Fig. 20. Use of vacuum cleaner to avoid the dispersion of contaminated powder.

5.6. Radioactive waste generated

Prior to the radiological characterization, 10 liters of contaminated liquids and sludge were removed from the retention tank and collected in a plastic container (Fig. 13). These effluents were contaminated with ^{137}Cs and ^{226}Ra. No more liquids or heterogeneous wastes were generated later.

During dismantling operations in the retention tank, solid radioactive wastes were generated: soil and concrete debris. These wastes were collected in plastic bags and measured with the contamination monitor MicroCont II (Fig. 21). The bags were placed in 200-litre drums, segregating soil from debris. Three drums with solid waste were generated. Some bags with soil measured less than 30 cps at the surface, and they were placed separately for further sampling and spectrometric analysis. The specific activity in this material may be shown to be below the clearance levels established in Cuban regulations [6], and consequently it is probable that it can be cleared and released.

Fig. 21. Monitoring of materials removed from the retention tank.

5.7. Repairing and testing of the retention tank

Repairing and maintenance operations included the valves and the retention tank itself. No information was available regarding the functioning (or the flowchart) of the canalization system. It was verified that in the compartment 3 (Fig. 9) there was a part of the outlet pipe and after that two valves (Fig. 22). The right valve had remained closed for many years. It was found that after the valve, the pipe was broken, and for this reason it was decided to lock off this part of the pipe (Fig. 22).

Fig. 22. Closing a ramification of the outlet pipe.

The walls and the floor of the retention tank were partially filled with concrete (10-12 cm layer on the floor). A smooth finishing was guaranteed with a cement mortar layer and epoxy painting (Fig. 23).

Fig. 23. Repairing the walls and floor of the retention tank.

The maintenance of the valves was a very important task in order to guarantee the adequate functioning of the canalization system (Fig. 24).

Fig. 24. Maintenance of the valves.

Once repair and maintenance operations were concluded, the canalization system was tested in order to verify adequate functioning. The retention tank was filled with water to a fixed level and the outlet valve remained closed for 10 days. After that period it was verified that the volume of water remained the same (Fig. 25).

Fig. 25. Testing the functioning of the canalization system.

5.8. Concluding remarks

- The procedure followed for radiological characterization of the retention tank at the Waste Processing Facility was adequate. The necessary information for planning and implementing D&D operation was obtained.
- The retention tank was decontaminated and repaired, guarantying an adequate collection and control of liquid effluents generated at the Waste Processing Facility.

6. Conclusions

The immediate dismantling and decontamination should normally be the most appropriate strategy for small facilities. Despite this fact, the reality shows that existing constraints, mainly associated with resources, have imposed deferred decommissioning on facilities.

The strategy selected for decommissioning should take into account regulatory, technical as well as financial considerations. Criteria for final release of a facility from regulatory control should consider the further use of the facility as well as the reconstruction operations needed before reuse. In some cases, low levels of radioactivity could remain in place with no significant hazard to the public and the environment.

In the case of a small nuclear programme and limited resources, international involvement and cooperation is needed for planning and conducting decommissioning projects.

The results have strengthened the Cuban national capabilities for conducting D&D activities and led to increased cooperation between D&D operators, regulators and users of radioactive materials.

7. Interactions with other Organizations and other CRP members

The Coordinated Research Meetings allowed the exchange of information about the experience of other CRP members in decommissioning projects. This experience has helped in the adequate planning and implementation of D&D projects in Cuba.

During the development of this CRP, specialists from CPHR have been involved in drafting of new IAEA documents and have attended IAEA Conferences [2] [3] related to decommissioning of nuclear facilities.

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The papers 2 and 3, as well as an annex contained in reference 4 were produced in the context of this CRP.

PROCESS OF SELECTION OF SUITABLE TECHNOLOGY FOR DECOMMISSIONING ACTIVITIES

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Abstract
The process of the selection of technology for decommissioning is one of the most important aspects of decommissioning. Selection methods have a large impact on the whole decommissioning process, e.g. radiation protection, radioactive waste (RAW) management, and on the cost of decommissioning. Radiation protection and economics are key factors for the selection of technology for decommissioning. There are also other important parameters — type of material, thickness, accessibility of technology, etc. NRI is equipped with technology for decommissioning and RAW management. The evaluation of the available technology has been made and areas for improving the technology have been identified. The NRI is being equipped with new equipment for segmentation and decontamination. Standard industrial technology with small modifications is being purchased and also special technologies for segmentation and decontamination are being developed with the assistance of external subcontractors. The process of the selection of suitable technology, its application, progress in remediation and lessons learned are described in the report.

1. Introduction

NRI is a leading institution in all areas of nuclear R&D in the Czech Republic. NRI has had a dominant position in the nuclear programme since it was established in 1955 as a state-owned research organization. In 1992, NRI was transformed into a joint-stock company.

The Institute's activities encompass nuclear physics, chemistry, nuclear power, experiments at a research reactor and many other fields. The main issues addressed at NRI in the past decades were concentrated on: research, development and services provided to the nuclear power plants operating WWER reactors, development of chemical technologies for the fuel cycle and irradiation services, and research and development in the industrial sector, agriculture, food processing and medicine.

After 50 years of activities in the nuclear field, there are many environmental liabilities that need to be remedied at NRI. NRI operates two research nuclear reactors, many other facilities such as a hot cell, research laboratories, technology for RAW management, radionuclide irradiators, an electron accelerator, etc.

There are three areas of remediation: (1) decommissioning old obsolete facilities (e.g. decay tanks, liquid RAW storage tanks, old RAW treatment technology, special sewage system), (2) processing of RAW resulting from the operation and dismantling of nuclear facilities and (3) elimination of spent fuel from nuclear research reactors. Remediation of the environmental liabilities started in 2003 and will be finished in 2012.

2. Description of work

Preparation of an inventory of contaminated equipment was the first step towards selecting the technology for decommissioning [1, 2]. The result was approximately 1500 m^3 of RAW weighing approximately 600 tons. This RAW comprises a research reactor vessel, a primary circuit, an evaporator, storage tanks, filters, piping, etc.

Radiation protection is the most important factor for selecting the technology for decommissioning. The level of contamination of RAW is up to tens of MBq/m^2, and the dose rate is up to hundreds of mGy/h. There are also other important parameters – type of material, thickness, accessibility, etc.

Further, an evaluation of the possible uses of various technologies for segmentation and decontamination with an aim to facilitate management of generated RAW and release into the environment was performed together with economic evaluation. Various methods were designed and tested. The results are described in the following chapters.

3. Selected technologies

NRI is equipped with the following technology for RAW management:

- Decontamination Centre.
- Facilities for RAW storage.
- Evaporation unit, cementation and bitumination unit.
- Laboratories for decontamination and RAW characterization.

The evaluation of the available technology has been made and areas for improving the technology have been identified. Standard industrial technology with small modifications is being purchased and also special technologies are being developed with the assistance of external subcontractors.

In Table 1 there is a list of methods used or considered for segmentation and decontamination. Besides the in-situ mechanical milling, all the technology is usually used in industry with small modifications.

Table 1. List of methods used or considered for segmentation and decontamination

Segmentation	**Decontamination**
Power hydraulic shears	Vacuuming (vacuum cleaner with HEPA filter)
Mechanical saw	High-pressure water jet
Nibbler	Chemical decontamination
Abrasive cutting wheel	Foam decontamination
Oxy acetylene cutting	Ultrasonic decontamination
Plasma arc cutting	Dry ice blasting
In-situ mechanical milling (segmentation of tanks, remote controlled)	Grit blasting (in-situ, in box) – considered
High pressure water jet cutting (considered)	

3.1. Segmentation

3.1.1. Power hydraulic shears

Two standard devices are used for shearing. The first is a mobile device. It serves for sheering material with a thickness of 2-3 mm (metal sheets), 2 mm (section steel) and 20 mm (tubes).

The second is a fixed device “Caiman” (Fig. 1) developed for iron scrap works. The length of the blades are 600 mm, maximum force 1550 kN and the devices can cut materials up to the following dimensions: round steel with diameter of 50 mm, steel girder with dimension of 180 mm. Shearing is a fast method, almost without production of secondary RAW.

Fig. 1. Power hydraulic shears.

3.1.2. Mechanical saw

This standard device is used for the segmentation of tubes and sections of steel (maximum diameter of 200 mm) and metal plates (maximum thickness of 5 mm). The device is cooled by oil emulsion. This method avoids the production of radioactive aerosols, but the character of secondary RAW is rather problematic (mixture of oil emulsion and filings).

3.1.3. Nibbler

A nibbler is a punch and die cutting tool that normally operates at a rapid reciprocation rate of the punch against the die, "nibbling" a small amount of sheet metal work piece with each stroke. The standard hand-held device is used – maximum thickness of material cut is 8 mm (stainless steel) or 10 mm (carbon steel), respectively. It is a fast method, avoiding production of radioactive aerosols. The character of secondary RAW is favourable (bigger chips). The thickness of material cut is limited.

3.1.4. Abrasive cutting wheel

Such standard tools are used for the segmentation of objects up to a diameter of 200 mm. It is a fast method, though producing radioactive aerosols and posing a fire hazard.

3.1.5. Oxy acetylene cutting, plasma arc cutting

Oxy-acetylene cutting is used for the segmentation of carbon steel parts with a thickness up to 15 – 20 mm. Plasma-arc cutting is used for the segmentation of carbon or stainless steel parts with a thickness of up to 30 mm (stainless steel) or 40 mm (carbon steel), respectively. It is a fast method for the segmentation of carbon or stainless steel, though producing radioactive aerosols and poses a fire hazard.

3.1.6. *In-situ mechanical milling*

Single-purpose remote controlled in-situ mechanical milling was a specially developed device which will be used mainly for the segmentation of tanks, but it can be also used for a planar object (Fig. 2). There are mainly two types of tanks made from structural steel with a thickness of 12 mm (bottom 14 mm) jacketed by stainless steel inside the vessel with a thickness of 2 mm and a volume of 10 m^3 (weight 4000 kg, length 3500 mm, diameter 2000 mm) or 63 m^3 (weight 9700 kg, length 9500 mm, diameter 3000 mm). The tanks are located within the bunkers and the space around is very limited (from 500 to 1000 mm), mainly under the tanks (from 300 to 600 mm). The basic parameters of in-situ mechanical milling are provided in Table 2.

Table 2. The basic parameters of the in-situ mechanical milling machine

Weight	100 kg
Diameter of segmented object	min. 1500 mm
Cutting rate	max. 250 mm/min
Kerf width	max. 16 mm

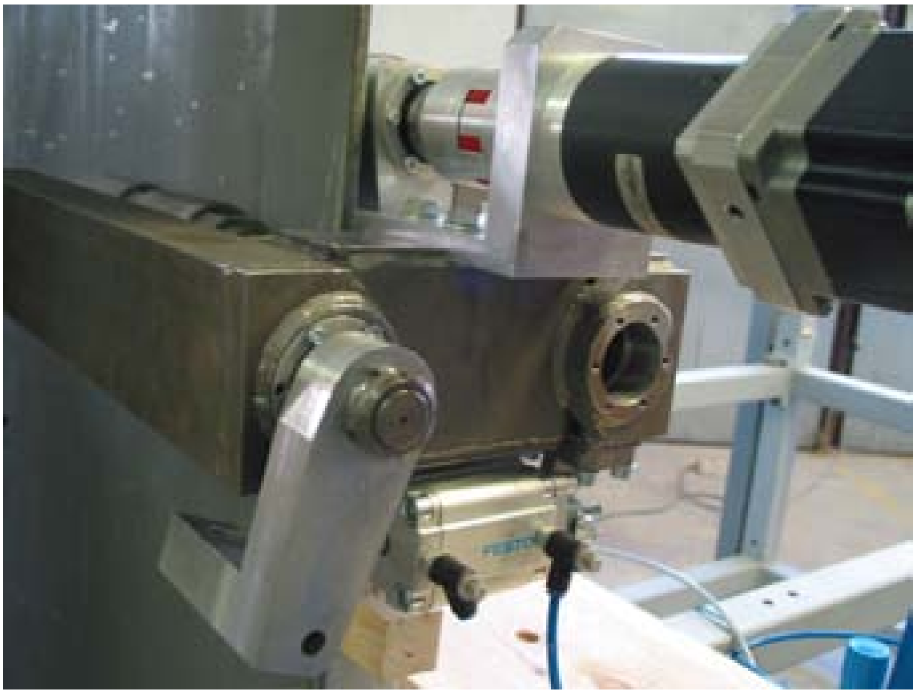

Fig. 2. Device for mechanical milling.

Sludge will be removed before segmentation and the tanks will be preliminarily decontaminated. The tanks will be segmented in rings with a width of 400 mm and then cut by hydraulic shears. The device will produce secondary RAW in form of small chips and almost no airborne contamination will be produced. The device has been successfully tested and will be used for segmentation in the near future.

3.1.7. *High pressure water jet cutting*

This technology could be used for the segmentation of equipment with a greater thickness (100 mm and more of steel) with respect to poor accessibility for other devices (mechanical segmentation).

Purchase of a standard industrial device was evaluated. However, after the evaluation it was decided not to purchase the device because:

- The price is rather high.
- Liquid RAW is produced by its application.
- There is a problem connected with the spread of contamination during application.

3.2. *Decontamination*

3.2.1. *Vacuuming (vacuum cleaner with HEPA filter)*

A special vacuum cleaner equipped with a HEPA filter is used for decontamination of building surfaces and equipment. The basic parameters are: maximum air flow 250 m^3/h (70 l/s), maximum negative pressure 30 kPa, and filtration efficiency 99,999 % > 0,3 microns.

3.2.2. *High-pressure water jet*

A standard device for high-pressure water jetting is used. It enables heating the water and adding the decontamination agents into the water. The basic parameters are: working pressure up to 200 bars, volumetric flow up to 980 l/h and temperature up to 140 °C.

3.2.3. *Chemical decontamination*

Chemical decontamination with use of solutions of acids (both organic and inorganic), lyes and other chemicals (detergents) are used for decontaminating contaminated surfaces (by washing or rubbing).

3.2.4. *Foam decontamination*

The foam is prepared by a foam generator. The foam contains detergents and other chemical agents. The foam can be in contact with the surface for a longer time, depending on the foam's stability. It limits the quantity of secondary RAW. The foam is very efficient for oily or greasy surfaces.

3.2.5. *Ultrasonic decontamination*

A decontamination ultrasonic bath with a volume of 1.3 m^3 (Fig. 3) is used for decontaminating segmented parts. The basic technical parameters are provided in Table 3.

Table 3. The basic parameters of ultrasonic bath

Inner dimensions	1400 x 1150 x 1000 mm
Active volume	1.3 m^3
Ultrasonic power	17 kW
Basket dimensions	1300 x 800 x 700 mm

Fig. 3. Ultrasonic bath.

3.2.6. Dry ice blasting

Dry ice blasting was developed as a safe, clean alternative to bead, grit, and sandblasting. Dry ice blasting has grown to become a vital part of the cleaning process in a remarkable variety of industries throughout the world. The cleaning process utilizes dry ice (solid CO_2) which is formed into 3 mm rice like pellets or blocks of dry ice which are ground into tiny particles the size of sugar crystals. These particles are then accelerated to supersonic speeds via a blasting unit and applied using a hand held or robotized blasting gun to the surface to be cleaned. Upon impact, the dry ice immediately turns from its solid state into carbon dioxide vapour expanding up to 540 times its volume. The energy released by the conversion of solid to vapour is considerable and is responsible for much of the cleaning process. The vapour disappears back into the atmosphere, leaving only the removed contaminant itself for disposal. The contaminant aerosols, if there are any, are filtered by standard air cleaning methods. Unlike conventional blast cleaning methods: grit, sand, plastic media, etc. Dry ice blast cleaning is non-abrasive to the impacted surface. Due to generating gaseous CO_2, it is necessary to use an efficient ventilation system. The decontamination procedure is shown in Figure 4.

Fig. 4. Decontamination by dry ice blasting.

Two devices have been purchased – Cold Jet Alpheus T-2 and Cold Jet Alpheus SDI-5 (Fig. 5). The big advantage is the very small production of secondary waste. Their parameters are provided in Table 4.

Table 4. Parameters of dry ice blasting devices

	Cold Jet Alpheus T-2	**Cold Jet Alpheus SDI-5**
Dry ice feed capacity [kg]	5.4	54.4
Supply air pressure range [bar]	2.4 – 12.1	2.8 – 17.2
Weight [kg]	47	195
Size (L x W x H) [cm]	56 x 36 x 51	94 x 62.2 x 114.3
Variable dry ice feed rate [kg/min]	0.2 – 0.7	0.5 – 2.7

Fig. 5. Cold Jet Alpheus SDI-5.

3.2.7. Grit blasting

Abrasive blasting is a very efficient method for decontaminating of material covered with a layer of corrosion, oils, grease, etc. The technology will be used for decontamination for release in the near future. The grit blasting will be used for both in-situ decontamination and decontamination in a special box.

4. Applications of selected technologies

The process of application is shown in the following examples. It is preferable to choose a simple method when standard or adapted equipment is used. The economical evaluation of the whole segmentation and decontamination process is necessary, and sometimes, it is better to dispose of the waste than to spend resources and time for segmentation, decontamination and measurement.

4.1. Decommissioning of an old evaporation system

An old evaporation system (Fig. 6) was used for the treatment of liquid waste. After it began to leak, it was put out of operation and, awaiting for decommissioning. The system consisted of the evaporator (boiler and separator), three drop separators and a condenser.

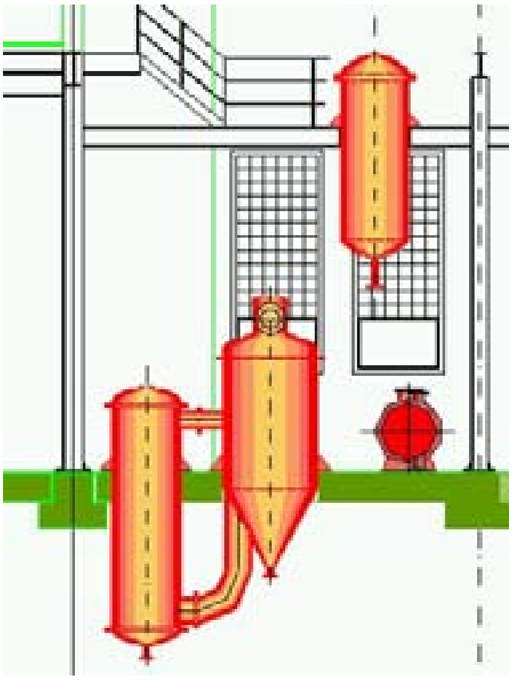

Fig. 6. Old evaporator system (schematic plan).

4.1.1. Segmentation of a condenser

The condenser was used for cooling vapours from evaporation. Dimensions of the condenser are as follows: diameter 0.7 m, length 4 m. The condenser is made from carbon steel (shell) and brass (heat exchange tubes). The process of segmentation is shown in Figure 7.

Fig. 7. Segmentation of a condenser.

Abrasive cutting wheel and hydraulic shears were used for segmentation. It was decided to treat and condition the RAW from the condenser segmentation for disposal because it would be difficult to decontaminate (strong corrosion of steel parts). It would be possible to decontaminate the brass tubes, but, with respect to their inner dimensions, it would not be possible to prove that the release levels were met.

4.1.2. Segmentation of drops separators

Three drops separators were used for separating water drops in vapours from evaporation. Dimensions of the drops separator are as follows: diameter 1.4 m, height 2.86 m. The separators were made from carbon steel (shell and internals) and contained special filtration material. The removal of the separator is shown in Figure 8.

Fig. 8. Separator removal.

An abrasive cutting wheel and oxyacetylene cutting were used for segmentation. It has been decided to treat and condition the RAW from the condenser segmentation for disposal because of the difficult decontamination foreseen (strong corrosion of steel parts).

4.1.3. Segmentation of the evaporator

The evaporator consisted of a heater and a separator (Fig. 9). The shell was made from carbon steel and heat exchanger tubes from brass. The heater was heated by steam; the evaporated vapour was separated in the separator. Dimensions of the evaporator were as follows: diameter 1.4 m, height 4.2 m (separator) and diameter 1 m, height 3.8 m (heater).

Fig. 9. Old evaporator (upper and lower parts).

Dismantling started with the removal of thermal insulation. The separator was partially dismantled by oxyacetylene cutting before removal because it was not possible to remove it as one piece (it was embedded in concrete) (Fig. 10).

Fig. 10. Removing the separator after partial segmentation.

The heater was removed in one piece (Fig. 11). The outer shell was dismantled by the nibbler to minimise airborne contamination. The tubes were dismantled by hydraulic shears.

Fig. 11. Heater before dismantling.

It has been decided to treat and condition the RAW from the evaporator segmentation for disposal because of difficult decontamination (strong corrosion of steel parts). It would be possible to decontaminate the brass tubes, but, with respect to its inner dimensions, it would not be possible to prove that the release levels were met.

Because the evaporator was contaminated also by alpha radionuclides (mainly ^{241}Am), overalls with masks supplied with fresh air were used for personal protection.

4.2. Pipe decontamination

The ultrasonic bath was used for decontaminating pipes from the decommissioning of the sewage system which served for transfer of liquid RAW. The system consisted of a stainless steel pipe network with a total length of 410 m, situated in an underground concrete corridor. The total amount of contaminated metal parts was approximately 20 metric tons.

A standard mechanical saw was used for the segmentation of pipes. The pipe parts such as joints and flanges and various corroded parts were sent for conditioning. A high-pressure water jet was used for internal and external preliminary decontamination of the pipes. Then, the ultrasonic bath with decontamination solutions was used. The decontamination was successful in most cases; some pipes were mechanically decontaminated by a special single-purpose instrument (an abrasive rotating device).

After decontamination, the contamination from outside was measured by a standard contamination instrument. Then the contamination inside the pipes (assumed to be maximum) was measured by a special tube detector and the parts with maximum value of contamination were cut off and then rolled out rather in the manner of an old “sea chart” for confirmation of the contamination measurement. Approximately 90 % of pipes were released into the environment.

4.3. Segmentation of a storage carousel

The carousel was used for storage of ionising radiation sources. The contaminated upper shielding lid made from cast iron with a thickness of 100 mm had to be cut before carousel decontamination. The use of flame cutting can lead to dispersion of contamination and internal contamination of construction material. Instead of this, a simple mechanical method was used for segmentation. In this way, the lid was cut with use of a standard drilling machine, making many holes one next to the other (Fig. 12). The lid was segmented into six parts, then decontaminated and released into the environment.

Fig. 12. Segmentation of carousel lid.

4.4. Decommissioning of storage tanks

Three steel cylindrical tanks (length 9.5 m, diameter 3 m, weight approximately 10 metric tons) each with a capacity of 63 m^3 located in underground bunkers served for receiving liquid RAW.

According to the original project, the remediation procedure would comprise decontaminating and dismantling the tanks. Then, new tanks for the storage of liquid RAW would be installed. A new concept has been prepared. The tanks will be decontaminated and after an investigation of their state, a polyethylene lining will be installed inside the tanks. In this way, the resources for segmentation and RAW processing and installation of new tanks will be saved.

The decontamination of the tanks is being performed now. The surface of one tank was contaminated by a bituminous product from the decontamination of a bituminisation unit by an organic solvent in the past. It was very difficult to remove this bituminous coating. At first, the application of the dry ice blasting was proposed but this could lead to spreading the contamination. Then, a method utilizing organic solvents was proposed but the amount of the solvent required could be rather greater. Additionally, the protection of workers could be demanding. Therefore, a very simple method was used – application of mineral oils to soften the bituminous coating. The coating was then removed manually and a very small amount of organic solvent was used to complete the decontamination (Fig. 13).

Fig. 13. Inner tank surface before and after decontamination.

4.5. Decontamination of a "semi-hot" cell

Dry ice blasting was used for decontamination of a "semi-hot" (i.e. hot cell containing lower levels of radioactivity) cell used for dissolving spent fuel in the past. The cell, made from cast iron, was partially decontaminated in the past and then its internal and external surface was covered with coating to prevent the contamination from spreading. The coating was partly removed by a paint remover at first and then dry ice blasting was applied for decontamination. Dry ice blasting was also used for decontaminating building surfaces – removal of contaminated coatings. Small amount of secondary RAW was generated and dispersion of contamination was limited. The "semi-hot" cell was then reused for other research activities.

4.6. Segmentation of drums for storage of spent fuel

Old EK-10 spent fuel was stored in NRI in special storage containers – 200 l drums filled with concrete (Fig. 14). Before the transportation of the SF to the Russian Federation for reprocessing, packing the SF was necessary. It was performed in a special hot cell.

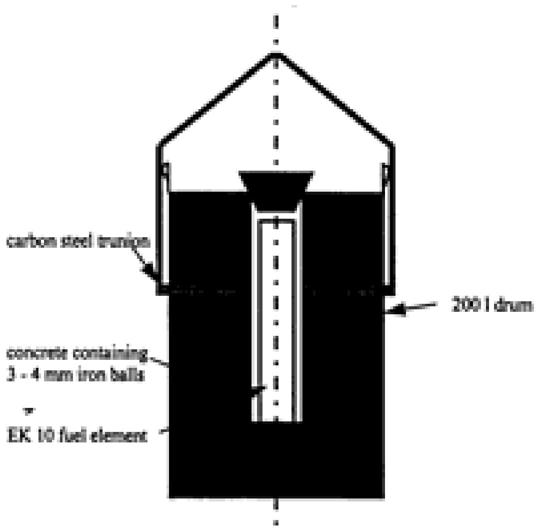

Fig. 14. Sketch of the storage container.

The drum was equipped with a steel trunnion that had to be removed before transferring the drum into the hot cell. Special manipulation equipment was developed together with the autonomous mechanical saw. Due to the fact that the dimensions of storage drums were often different, the developed technology did not work properly and intervention of the workers was necessary. Then, the obtained dose to the worker increased and a new procedure had to be developed. Instead of the mechanical saw, a mechanically remote controlled plasma arc cutting was used (Fig. 15). The procedure was very simple and quick and intervention by the workers was not necessary.

Fig. 15. Plasma arc cutting.

4.7. Decay tanks

The building 211/5 Decay tanks had been in use since 1961. The building was designed for storage and decay of concentrated short-lived RAW, but also RAW containing long-lived radionuclides was shipped there. The building is submerged in the terrain on three sides (Fig. 16). It contains two cylindrical tanks (length 9.5 m, diameter 3 m, weight approximately 10 metric tons), each with a capacity of 63 m^3 (Fig. 17). The decay tanks are made from structural steel jacketed by stainless steel inside the vessel. They are placed into two separate concrete bunkers located partially below ground. Above the bunkers, a building with tank inlet pipes and ventilation equipment are located.

Fig. 16. Uncovered bunkers.

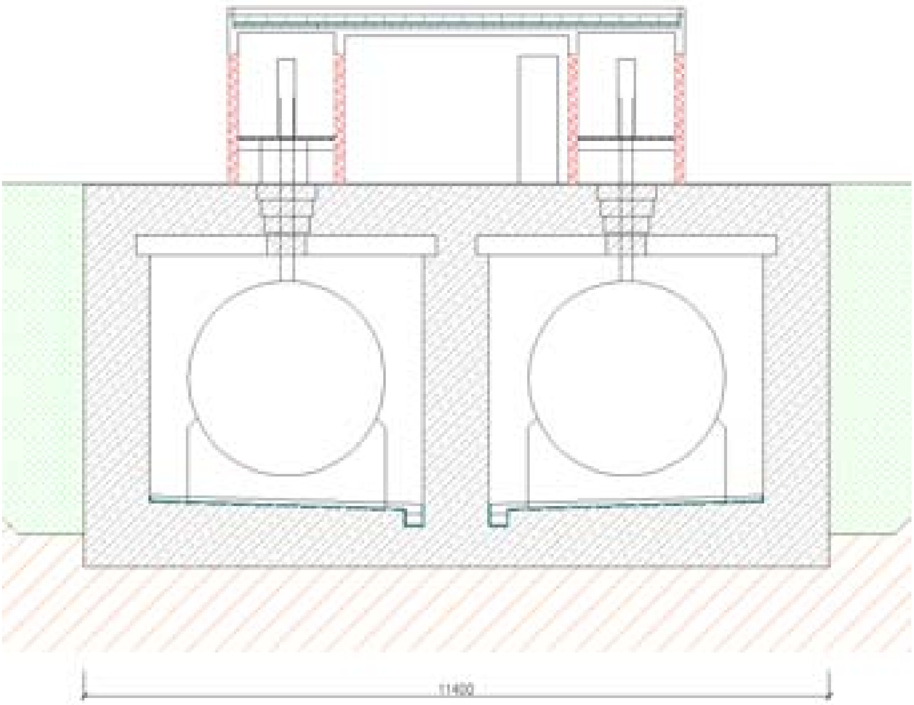

Fig. 17. Decay tanks (section).

The tanks contain not only liquid RAW, but tank B contains also solid RAW (Fig. 18). The main identified radioisotopes are ^{60}Co and ^{137}Cs. However, the presence of ^{239}Pu is also assumed. Solid RAW consist of tins with irradiated metallic samples and residues of spent fuel. The maximum dose rate is above the pile of solid RAW (hundreds of mGy/h). The leakage from the tanks and direct irradiation from in-situ material were identified as the main risks to the environment and/or to employees.

Fig. 18. RAW stored in the tank B.

The remediation procedure will be as follows:

(1) A hall above the decay tanks has been built (Fig. 19).
(2) The old above-ground building was demolished. An industrial concrete saw was used for segmentation of the concrete inlets. The dismantling of the concrete structure was done by hydraulic devices (Fig. 20).
(3) A special remote controlled manipulator will be installed into the tank inlet. The control room of the manipulator will be placed in front of the bunker.
(4) The liquid RAW from the tanks will be removed. The liquid from tank A (with lower activity) will be transported via a special tank to the liquid RAW processing. The liquid from tank B (with higher activity) will be cemented onsite with the specially developed cementation unit.
(5) Solid RAW will be removed to a special shielding container and transported to a special hot cell facility for processing. Then RAW packed in special cases will be loaded into disposal units and sent for disposal.
(6) The tanks will be decontaminated by a high-pressure water jet and abrasive blasting. The tanks will be dismantled by special segmentation equipment. The RAW will be either released into the environment or disposed. The building will be decontaminated for unrestricted use.

The construction of the facility was finished in 2007; removal and processing of RAW will start in 2008. Decontamination and segmentation of the tanks will be carried out in 2009 and then the building will be decontaminated.

Fig. 19. New hall.

Fig. 20. Segmentation of tank inlets.

5. Interactions with other CRP members

Use of the nibbler for segmenting the tanks was discussed with Mr. Dadoumont, SCK-CEN, Belgium, in 2006. SCK-CEN has considerable experience with the use of this device. On the basis of these discussions and recommendations, one nibbler has been purchased and is successfully used for segmentation of contaminated equipment.

Experience with decontaminating the inner surfaces of the tubes by rotating devices was discussed with Mr. Fabbri, CNEA, Argentina. This equipment was used for decontaminating the tubes, when the ultrasound decontamination was not successful.

Experience with using of molten salts in the field of radioactive waste management was discussed with Mr. Lainetti, CNEN, Brazil.

6. Conclusions

On the basis of this analysis, the optimal technology for decommissioning old environmental liabilities has been proposed. Special technologies for segmentation and decontamination are being developed with the assistance of external subcontractors. Further, the use of the technology will be economically advantageous.

NRI has gained a great deal of experience in the field of RAW management and decommissioning of nuclear facilities and will use its facilities, experienced staff and relevant data for the successful realization of the remediation.

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EXPERIENCE GAINED DURING THE DECOMMISSIONING OF DANISH RESEARCH REACTORS DR 1 AND DR 2
INNOVATIVE AND ADAPTIVE TECHNOLOGIES IN DECOMMISSIONING OF NUCLEAR FACILITIES (TS.40.07)

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Abstract

The two research reactors DR 1 and DR 2 have been the first out of five facilities at Risø National Laboratory in Denmark to be decommissioned by Danish Decommissioning. They thus have a particular potential for posing challenges and unforeseen problems that would necessitate the use of special equipment and methods. The two nuclear facilities that were expected to give the smallest problems, DR 1 (only 2 kW) and DR 2 (5 MW, but the reactor has been closed for more than 25 years), should serve as test beds prior to the decommissioning of the other nuclear facilities, in particular the reactor DR 3 (10 MW, shut down in 2000). The decommissioning of DR 1 and DR 2 was planned to take place from 2004 – mid 2009. Decommissioning of all facilities at the Risø site is planned to run through 2018. The overall objective of the decommissioning at the Risø sites is to reach "green field" so that the area and possible remaining buildings can be used for other purposes without any restrictions. To a large extent, dismantling works have, therefore, been carried out by use of tools and skills already available in DD. However, a number of special tools have been acquired or considered for specific operations and works. This paper focuses on the methods and special tools acquired or being considered for the decommissioning of the DR 1 and DR 2.

Introduction

DR 1 was a 2 kW thermal homogeneous solution-type research reactor which used 20% enriched uranium as fuel and light water as moderator. The first criticality was obtained in 1957. The reactor core consisted of a spherical vessel with an outer diameter of 31.75 cm containing a solution of uranyl sulphate in light water. The core was positioned at the centre of a cylindrical reflector consisting of graphite bars stacked in a steel tank, which was placed inside a shield of magnetite concrete. When the reactor was in operation hydrogen and oxygen were produced by radiolysis of the core solution. In order to recombine the produced gases to water vapour, the reactor was provided with a recombiner tank.

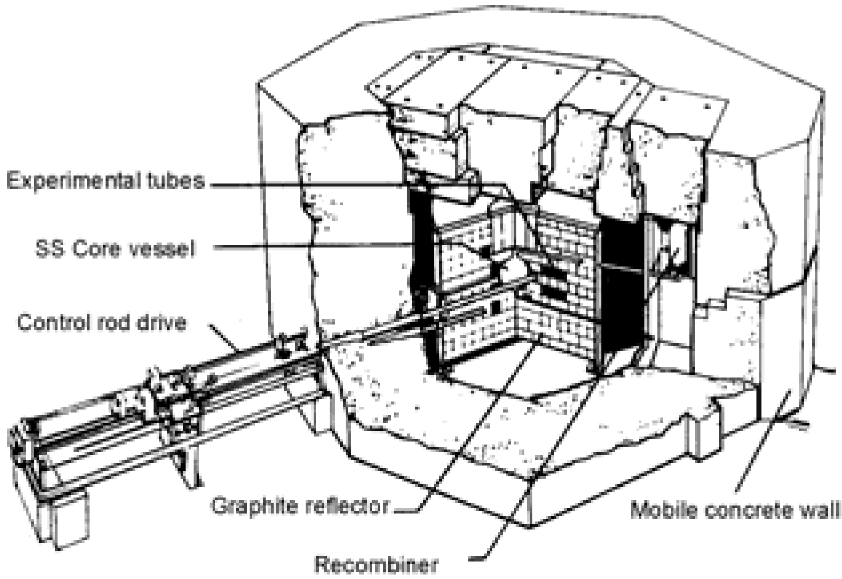

Fig. 1 Sketch of DR 1.

The plan for decommissioning of DR 1 was produced during 2003 and approved by the nuclear authorities during spring 2004. In the meantime, detailed work plans for the removal and demolition of active components had been set up, and the actual decommissioning work could start immediately after approval of the decommissioning plan. The dominating part of the remaining activity was concentrated in two components, the reactor vessel (the fuel solution had been removed) and the recombiner (Fig. 1). Although not extremely active, these components required some degree of remote handling and, of course, detailed planning of all operations and consideration of the risks involved.

DR 2 was a tank-type, light-water moderated and cooled reactor with a power level of 5 MW (Fig. 2). It went critical in 1958. The reactor was finally closed down in 1975 and later partially decommissioned. Prior to re-start of decommissioning in 2006, the reactor block and the primary coolant circuit remained as the major tasks. The reactor tank was made of aluminium. It had a wall thickness of 9.5 mm, a diameter of 201 cm and a height of 808 cm.

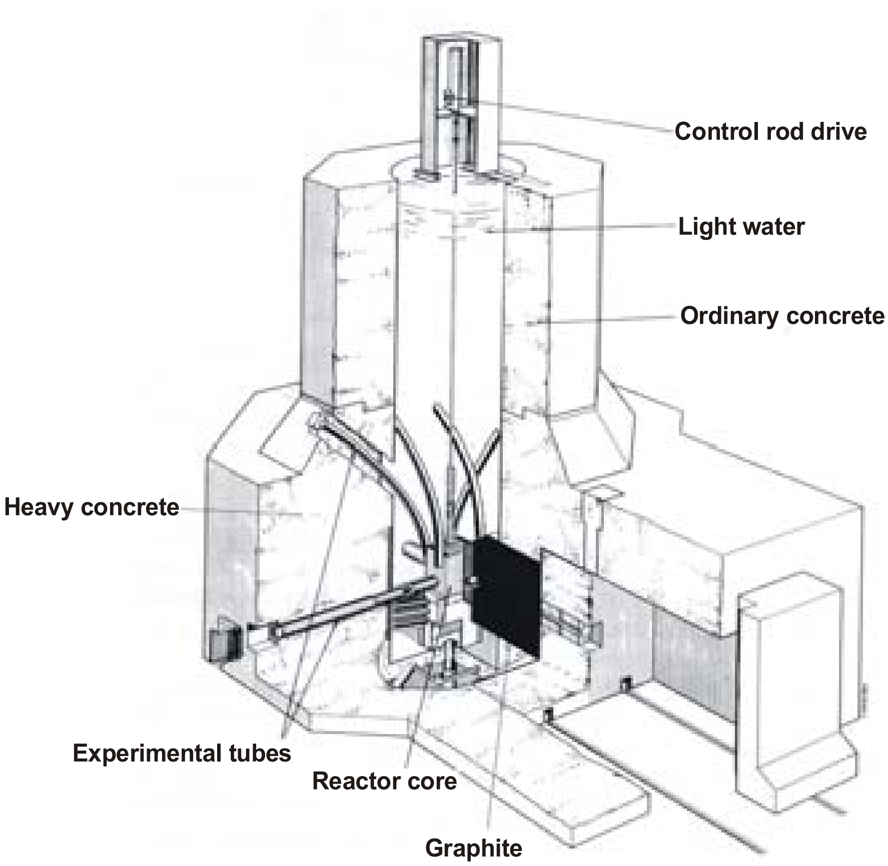

Fig. 2 Cut-away drawing of DR 2.

More extensive descriptions of the reactors and other nuclear facilities at the Risø site can be found in [1]. The planning of decommissioning of the reactor commenced during spring 2006, based on the information acquired during a characterisation project carried out during 2002-03 [2].

1 Decommissioning of the DR 1 facility

As mentioned above, the major part of the activity was concentrated in the reactor vessel, the recombiner, and the 2⅜" pipe connecting the two. All components are of stainless steel. Although not extremely active, these components required some degree of remote handling and, of course, detailed planning of all operations and consideration of the risks involved in order to minimize personnel doses and occupational injuries. A detailed account of the decommissioning works can be found in the final report of the project [3].

1.1 Decommissioning work performed

The permit from the authorities[1] to commence actual decommissioning work was given in October 2004, and active work started during October. Before then, external non-active or only slightly active systems had been dismantled, such as the control rods and their drive mechanism and the secondary part of the cooling system. Dismantling the latter was delayed somewhat, as it turned out that the insulation of the pipes contained asbestos. This required the contracting of an authorised company to remove the asbestos, observing all the safety precautions prescribed for this type of work.

The recombiner was taken out during October, and the graphite reflector and core vessel were removed during January 2005. Subsequently, the steel reflector tank was lifted out and cut into pieces of approximately 50×80 cm. More details about the dismantling methods follow in sections 1.4 to 1.7.

1.2 Decommissioning approach and tools used

The overall objective of the decommissioning at the Risø site is to reach "green field" so that the area and possible remaining buildings can be used for other purposes without any radiological restrictions. Risø National Laboratory wished to re-use the DR 1 building for other purposes. Therefore, Danish Decommissioning undertook to remove the reactor block and clean the building to a state where building clearance levels can be met.

As mentioned at the beginning of this chapter, there were essentially only three components that could cause any concern with respect to radiation, namely the reactor vessel, the recombiner and the pipe connecting the two. Danish Decommissioning, therefore, chose the approach to remove these components as early as possible in order to reduce radiation levels for the remaining work.

1.3 Own staff or contractors?

As mentioned earlier, Danish Decommissioning has decided that as much as possible of the decommissioning work should be performed by DD's own technical staff, many of whom have a long experience from the operation of the facilities. They know the facilities and are experienced in radiation work. For DR 1, the majority of the dismantling work was carried out by two technicians, generally supervised by either the project leader or his deputy. For some operations one or two additional technicians assisted. DD's health physics staff supervised all operations.

For the demolishing of the biological shield, DD chose to let an external contractor do the work instead of acquiring equipment and educating our own staff. The presence of external staff required some extra instruction and supervision of the work in order to ascertain that the rules for work in classified areas were adhered to, and that no material was taken outside the fence without having been cleared.

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[1] The National Institute of Radiation Protection and the Nuclear Division in the Danish Emergency Management Agency

1.4 Methods of dismantling

Selection of decommissioning methods for DR 1 started when the first overall plan was drafted for decommissioning of all nuclear facilities at the site. At this point, rough ideas about how to take apart the reactor were sketched and the required effort was assessed. A somewhat more detailed planning was made in the project description put forward for approval by the nuclear regulatory authorities. But the selection of precise approaches and tools to be used in the individual decommissioning operations to a large extent was made only during the detailed preparation of these operations.

In general, existing or off-the-shelf tools could be used, since there was no need for robots or other sophisticated remote handling equipment. In some cases special tools or modifications of existing tools were made in our own workshop.

1.5 Dismantling the recombiner

Contact dose rates at the surface of the recombiner were of the order of 5 mSv/h. Brief contacts with hands in order to place tools or lifting equipment was by no means excluded. Spectrometric measurements had shown that the source of the γ-activity was ^{137}Cs.

Fig. 3 shows the recombiner seen from above. Its outside diameter is 270 mm and the height ~500 mm. It weighs 30 kg. At the bottom, a flange connected it to the pipe leading to the core vessel (Fig. 4). The recombiner rested on four feet that were bolted to two beams below, as can be seen in Fig. 3. A number of cooling pipes and cables for measuring equipment and power supply were attached to the recombiner.

Fig. 3 The recombiner seen from above.

Fig. 4 Flange at the bottom of the recombiner.

In the initial planning it was contemplated to cut the connecting pipe between the recombiner and the core vessel by means of a hydraulic tool, which DD already had. The tool is able to seal the two ends cut away by pressing them before cutting in the middle. In this way the risk of releasing possible contamination would be minimised. However, test cuts on similar piping showed that the tool probably would not be powerful enough to press and cut the 2⅜" stainless steel pipe. Since a larger tool would be very expensive and since radiation levels were moderate, it was considered justifiable to disconnect the recombiner by opening the flange at the bottom and quickly replacing the two open ends with blind flanges. This operation was carried out without any particular problems; the bolts and nuts came apart fairly easily. In order to reduce doses to the technicians, extension-shafts were used for the spanners. No loose contamination escaped during the dismantling; but smear tests confirmed that there was a layer containing ^{137}Cs at the inside of the pipe and flange. Whole-body doses to the two technicians who carried out the work were measured to 102 and 30 microSv respectively. Doses measured with finger dosimeters were 500/600 and 150/100 microSv to fingers on left/right hand for the two.

The hydraulic cutting tool served well, however, for cutting smaller pipes, such as the one seen in the lower part of Fig. 3. Fig. 5 shows the tool in action. It can be operated remotely in cases where it can be lowered, hanging in the hydraulic hose. The result of the cut can be seen in Fig. 6.

Very small diameter pipes (≤ ¼") were cut with an ordinary wire cutter, as were power- and signal cables.

Fig. 5 Hydraulic cutting tool in action.

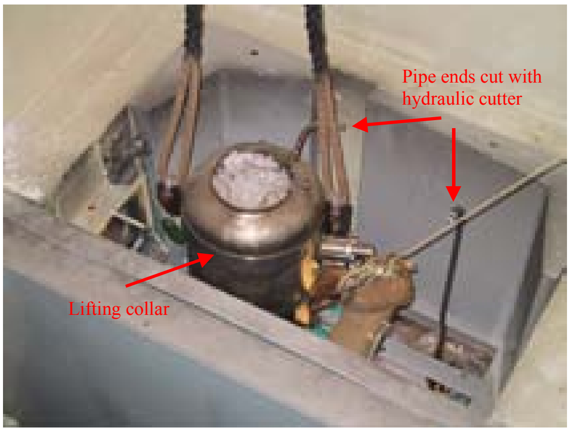

Fig. 6 The recombiner ready for lifting.

For lifting out the recombiner in order to transfer it to a waste drum, it was first considered to simply attach lifting gear to some of the pipes protruding from the component. This probably would have worked well, but in order to ascertain that there would be no risk of dropping it during the transfer, a simple collar was made, which could be mounted around the recombiner in a couple of minutes (Fig. 6).

The recombiner was lifted out and placed in a shielded drum that was transferred to a shielded cell made of concrete blocks in a corner of the reactor hall. The lifting gear seen in Fig. 6 was disposed of together with the recombiner in order to save doses to the personnel.

1.6 Dismantling the reflector and the core vessel

As can be seen from Figure 1, the core vessel was sitting in the middle of a tank, surrounded by the reflector. The reflector consisted of more than 300 graphite bars measuring 10×10 cm in cross section and having varying lengths up to 126 cm. There were 13 layers, and the direction of the bars in each layer was perpendicular to that of the layers above and below. Fig. 7 shows the top layer.

Fig. 7 Reflector tank (and recombiner compartment to the right).

There were three pipes connected to the core vessel which had to be removed before the vessel could be pulled out: the 2⅜" pipe leading to the recombiner, a ¼" drainpipe going from the bottom of the vessel and out through the reflector to the space below the recombiner (Fig. 1), and a 1" aluminium pipe going through a steel pipe that was welded into the core vessel. The aluminium pipe served to place small items in the centre of the reactor core for irradiation. This pipe could be pulled out fairly easily.

1.6.1 Disconnecting the drainpipe from the core vessel

It was evident that in order to remove the core vessel, at least seven layers of graphite would have to be taken out and that on the way, the connecting pipe to the recombiner could be cut loose. In order to minimize radiation doses to the personnel we wanted to find a way to disconnect the drainpipe at the bottom of the vessel without having to remove further layers of graphite. A number of possible approaches were considered:

(1) Entering cutting tools through the control rod channels and cutting the drainpipe just below the core vessel.
(2) Drilling down through the core vessel (and the 2⅜" pipe coming out at the top) and drilling away the drainpipe in the bottom.
(3) Drilling down through the graphite along the side of the core vessel after cutting the 2⅜" pipe away.
(4) Drilling horizontally from the outside of the biological shield hitting the drainpipe below the core vessel.
(5) Cutting the drainpipe where it protrudes from the reflector tank and pulling it (~75 cm) out together with the core vessel.
(6) Drilling a 50 mm core through the graphite around the drainpipe, entering from the recombiner vault.

Advantages and disadvantages by the alternatives are summarized in Table 1.

Table 1. Assessment of alternative methods to cut the drainpipe

Nº	Advantages	Disadvantages
1	Could be done without any personnel doses. Attractive "smart" method.	Space was very restricted for entering tools and it was not possible to find the drainpipe when searching with an endoscope.
2	Could be performed at some distance using a special set-up. Low personnel doses.	Cooling coils inside the core vessel might be in the way. Requires the mounting of a stiff structure at the top of the reflector tank to keep the drill in line.
3	Could be performed at some distance and the active components could be shielded during the operation.	Setting up the drilling equipment will have to take place after six or seven layers of graphite have been taken out and will require the mounting of a stiff structure for the drilling machine.
4	Could be done without any personnel doses.	A long distance to drill (~225 cm) through concrete, steel and graphite. Risk of missing the target.
5	Easy approach.	Risk of failure (pipe stuck and potential damage to the core vessel). Considered too uncertain.
6	Relatively easy approach. Deviations from the correct direction could be monitored on the way by watching the cores taken out (~each 20 cm). Required drill tooling available.	Radiation doses to technicians during mounting of equipment and during drilling. Spreading of graphite dust outside the reflector tank.

In the end, alternative 6 was selected. The drilling could be carried out by means of already existing equipment. A frame was welded to the wall of the reflector tank for mounting of the drilling machine. Fig. 8 shows the set-up at the start of the drilling. As far as possible, the graphite dust generated was caught at the entrance hole by a vacuum cleaner; but some (slightly active) dust found its way to the floor outside and caused some contamination. Nevertheless, the operation was considered very successful; the drill followed the correct direction and the drainpipe was cut exactly where it was supposed to be cut.

Fig. 8 Drilling a core around the drainpipe

1.6.2 Removing the reflector and the core vessel

The characterisation project that had been carried out prior to the decommissioning itself indicated that the radiation level around the core vessel would be of the same order of magnitude as around the recombiner, i.e. ~5 mSv/h. The level at the top of the reflector was measured to 50-150 microSv/h before removal of the graphite started. It was, therefore, desirable to remove the graphite elements by some kind of remote handling.

Since the graphite elements had a very smooth surface and were not too heavy (the longest ones weighed around 23 kg) it was decided to investigate the possibility for using suction pads for lifting out the elements. Three suction pads each with a diameter of 75 mm were bought and mounted on a beam as shown in Fig. 9. Tests carried out in the workshop showed that the lifting capacity of the three was at least 75 kg. In principle, one suction pad should be able to lift the heaviest of the graphite bars. We, therefore, decided to use this approach.

Two technicians operated the system: one stood at the top of the biological shield and manoeuvred the beam with suction pads into position by means of a long rod, and the other operated the air supply for the suction pads and the swinging crane used for lifting out the graphite elements and transferring them to the roof of the control rod house (Fig. 10, 11). Here each element was weighed, measured and registered. This kept two other people busy in addition to a health physics technician who surveyed the working environment and the elements into active and potentially non-active that could be free-released. As it turned out, there seemed to be activity levels above the perceived clearance levels in most of the elements (at the time of taking out the elements clearance levels had not yet been issued by the authorities) performed measurements of the radiation from each element. Finally, the graphite elements were transferred to small containers for transfer to the intermediate storage for radioactive waste and eventually to a repository that has not yet been established in Denmark. Meticulous registration was carried out with a view to being able to sort the waste effectively.

Fig. 9 Beam with suction pads.

Fig. 10 Beam lifting demonstration.

Fig. 11 Lifting preparation.

After the removal of seven layers of graphite and parts of the eighth layer, the 2⅜" pipe could be cut loose from the core vessel and a mounting could be fixed for lifting out the core vessel. After the removal of another layer, the vessel could be lifted out and transferred to a shielded waste drum (Fig. 12).

Although the radiation level had decreased in the reflector tank after the removal of the core vessel, also the remaining layers of graphite were taken out by means of the suction pads and manoeuvred from the top of the biological shield. However, now it was more acceptable that the technician stepped down into the tank to force free, graphite bars that were stuck together — this was the case in particular for the first bar to be taken out from a layer. Fortunately, it was mainly in the lower layers that this problem occurred.

In general, the lifting out of the graphite elements went surprisingly smooth. On the average it took about 1½ hour to take up one layer, consisting of about 30 individual bars.

Fig. 12 The core vessel lifted out.

1.7 Removal and cutting of the reflector tank

After being emptied and cleaned, the reflector tank was lifted out and placed in an open space in the reactor hall. It is somewhat activated and has to be deposited as radioactive waste. In order to reduce volume, it was cut into smaller pieces. As DD already had a nibbler that could cut this thickness of plate (6 mm), this method was chosen because it does not produce any sparks or dust (Fig. 13). Only the top flange had to be cut with a right-angle grinder. The cutting went very quickly, and the shroud of the tank was cut into pieces of ~50×80 cm. The bottom, which is somewhat thicker, was stored in one piece until further.

Fig. 13 The nibbler and the half-cut reflector tank.

1.8 Summary and conclusion

DR 1 was a small reactor without very high activity contents. Therefore, demolishing it was not a very demanding task in itself. But as this was the first decommissioning task in Denmark, much focus was put on doing the work without any incidents on the way. Furthermore, since two more complicated reactors and a hot cell facility await to be decommissioned in the future, it was also part of the DR 1 task to examine methods and tools.

In general, the dismantling could be carried out by means of tools and equipment that DD already had or that could be made in DD's own workshop. In particular, the use of suction pads for taking out the graphite reflector was a cheap way of reducing personnel doses. Table 2 gives a summary of the tools and equipment used.

Table 2. Summary of particular tools and equipment used during dismantling of reactor systems

Tool/Equipment	Used for	Existing or new. Bought or made in the workshop.
Hydraulic pressing- and cutting tool	Recombiner piping	Existing
Extension-shafts for spanners	Recombiner bolts and nuts	Existing/bought
Wire cutter	Electrical wires and small-diameter piping (≤ ¼")	Existing
Collar for lifting recombiner	Transfer of recombiner to waste drum	Made in workshop
Core-drilling tool	Drilling out the drainpipe	Existing
Suction pads	Lifting out graphite bars	Bought
Swinging crane	Lifting out graphite bars	Bought
Nibbler	Cutting up the reflector tank	Existing
Endoscope	Inspecting core-vessel internals	Existing

All choices of tools and methods were made in close cooperation between project leader, project engineer, technicians and health physics staff. This was a very fruitful approach, securing the utilisation of all relevant skills and knowledge and giving a common understanding of the tasks and their performance.

2 Decommissioning of the DR 2 facility

DR 2 is the second of three research reactors at Risø National Laboratory in Denmark being decommissioned. Dismantling work began in the spring of 2006 following a two-year planning period. DR 2 was a 5 MW pool-type light water cooled and moderated reactor, the Thermal flux is 5×10^{13} n/(cm^{2}s) (max). The reactor was in operation from 1959 to 1975. The reactor thus has had more than 30 years of decay time; but it still contained components with a substantial activity. The decommissioning of the DR 2 commenced properly in May 2006 and is expected to be completed in 2008.

2.1 Decommissioning approach and tools used

Selection of decommissioning methods started when the first overall plan was drafted for decommissioning of all nuclear facilities at the site. A much more detailed planning was made in the project description put forward for approval by the nuclear regulatory authorities and when setting up the budget to be approved by the Parliament's Finance Committee. But the selection of precise approaches and tools to be used in the individual decommissioning operations to some extent is being made during the detailed preparation of these operations. The general approach of Danish Decommissioning is to do as much of the dismantling of active components as possible with its own staff and only to call in external contractors for work that involves little or no radioactivity. One consequence of this approach was the plan to acquire a rather expensive wire cutting tool for the dismantling of one of the more active parts of the reactor internals. However, this plan was abandoned because skilled external contractors probably would be able to do the work in question better and faster than DD's own staff. This section will touch upon this subject and give an overview of the considerations behind the selection of other major tools and dismantling methods.

2.1.1 Special tools acquired or being considered

To a large extent the decommissioning works on the DR 2 has been carried out by use of tools and skills already available in DD.

However, a number of special tools have been acquired or considered for specific operations and works. In the presentation at the CRP meeting in Keswick, UK, in 2006 the considerations and conclusion on the following tools was included.

- Hydraulic cutter for cutting aluminium tubes (“S, R, T”);
- Automatic band saw for separating radioactive and potentially non-radioactive parts of larger tubes ("B-tubes"); and
- Hydraulic press for reducing the volume of tubes

In general, the tools served their purpose. However, each tool has its advantages and disadvantages; some limitations in use was also found. The findings and conclusions for each tool can be found in the table shown in annex 1.

For the removal of particular structures the following special tools were acquired or being considered:

- Removal of graphite from the thermal column by use of pneumatic lifting devices;

- Plasma cutter for cutting and downsizing metal structures (primarily steel and aluminium); and
- Dismantling of the concrete reactor shield by wire saw, hydraulic splitting or hydraulic hammer

The findings and conclusion for these tools are given in more details in the following sections.

2.2 Removal of graphite from the thermal column by use of pneumatic lifting device

The thermal column contained approximately 200 units of graphite stringers of mainly 1 meter length with a total weight of around 2 tons. It was necessary to remove the graphite to allow access to the lead nose of the column (Fig. 14).

The graphite could be approached on the outer side for measuring of radiation level but there were concerns that the inner part of the stringers would expose serious levels of radiation. It was therefore decided that close handling by the staff should be minimised to the extent possible.

In relation to other works, DD-staff have developed special pneumatic tools with vacuum suction pads and methods for lifting and removal of graphite and other materials (Fig. 9). The device was therefore already available in DD and required only minor maintenance and changes before it could be applied (Fig. 15).

The work included the removal of all the graphite stringers to dedicated containers including measuring and registration of each individual stringer. This registration together with experiments carried out on selected stringers would serve to identify stringers that had to be annealed for Wigner energy before final storage.

The pneumatic tools were mounted on an aluminium bearing beam for lifting the graphite stringers. An extension arm was also mounted with a pneumatic device for pulling the stringers out of the thermal column. The removal of the stringers by use of the pneumatic devices proved efficient and the work was carried out safely and according to the work plan drawn up (Fig. 16).

Fig. 14 Thermal column with graphite.

Fig. 15 Graphite stringers removed by use of vacuum lifting device.

Fig. 16 Arrangement for remote handling of graphite and packaging of container.

2.3 Plasma cutter for cutting and downsizing metal structures

The structures inside the reactor tank were all made of aluminium, including the thermal column and the grid plate and its bearing ‘legs’.

The radiation level measured from the lead nose (the protruding portion of the thermal column), after the removal of the graphite stringers was around 2 mSv/h. From previous measurements and investigations it was known that the highest radiation level inside the reactor tank was right in front of the lead nose with a level of 60 mSv/h. The majority of the radiation was considered to be coming from the grid plate but a fair amount must be expected to be coming from the lead nose with its original placing directly up to the core of the reactor. From camera and video inspection of the structures inside the reactor tank it was deemed necessary to remove the lead nose before the grid plate could be accessed properly and removed.

After removal of the graphite it was found that the majority of the radiation in the thermal column could be located to the central part of the lead nose containing built-in thermo elements used for monitoring purposes during the operation of the reactor (Fig. 17). It was possible to shield the majority of the radiation from here by using standard 300 mm concrete shielding blocks (heavy concrete). This allowed manual cutting of the lead nose if a tool could be found that was capable of cutting and at the same time reasonably fast.

According to the documentation available from the construction of the reactor, the thermal column was made of 19,mm aluminium plates lined with 6 mm Boral plates on the inside towards the graphite. A number of tests were made (on inactive material outside the reactor) with various cutting tools, such as circular saws of different types and plasma cutter. The problem was to cut through a construction consisting of both the ‘soft’ material aluminium and the particularly ‘hard’ Boral plates, with a thickness totalling 25,mm. It should be mentioned that the plasma cutting requires pressured air of minimum 6 bars. This is available in the DR 2 building as a basic supply.

The various types of saws proved to be inefficient and difficult to handle and in particular they did not turn out to be as fast as preferred. On the contrary, the plasma cutter model tested proved to be the solution, capable of cutting directly through the structure. Also, the plasma cutter proved to be light in weight and easy to manoeuvre. In addition, the plasma cutter could easily be remotely controlled.

The cutting of the lead nose from the thermal column was performed by mounting the plasma cutter to an extension arm of 2 meter. The cutting was performed by 2 members of the DD-staff, with one doing the cutting with the extension arm and the second one managing the power switch from a distance of approximately 4 meters to the column. This was found to be securing maximum safety in relation to the radiation but also to the fact that the plasma cutter requires high voltage supply and works by electrical contact between the cutter and the medium to be cut (Fig. 18).

Fig. 17 Thermal column with lead nose exposed.

Fig. 18 Plasma cutting of the lead nose.

It was finally concluded that the plasma cutter for cutting and downsizing structures of steel and aluminium (in steel: up to 28 mm fine cut and up to 40 mm rough cut), was necessary for this task at DR 2 and, it had been as useful as expected and was worth buying. The plasma cutter can also be used in other projects with the primary advances to be:

- Fast cutting and also cuts in Boral;
- Light weight; and
- Hand held and easy to mount to an extension arm.

The plasma cutter was subsequently used to cut out the grid plate and other remaining structures in the reactor tank, which were lifted out by crane (Fig. 19, 20).

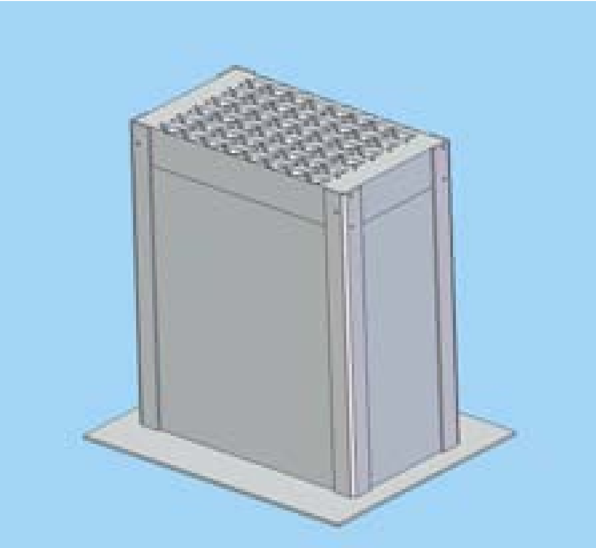

Fig. 19 Grid plate of aluminium, sketch.

Fig. 20 Grid plate hanging in crane.

2.4 Dismantling of the concrete reactor shield by wire saw, hydraulic splitting or hydraulic hammer

The demolition of the concrete reactor shield was to be carried out by an external contractor. After prequalification, four selected demolition contractors were invited for bidding on the work. The main requirements to the competition of the bidding and the demolition contract included:

- Method to be proposed by the bidder (e.g. wire saw, hydraulic splitting or hydraulic hammer);
- Only by use of dry methods (no water!);
- Use of tent and no spreading of contamination to the building and the environment (tent maintained under negative pressure);
- Concrete and other materials from the reactor shield to be separated in active and non-active materials;
- Optimal filling of containers with active waste for deposit;
- Maximum safety and health protection; and
- Full documentation and Quality Assurance.

After selection and contracting with the best bidder the following method of demolition and tooling was agreed:

- Demolition of concrete by use of:
  - Hydraulic hammer by use of a 'Brokk' demolition robot, remote operated
  - Dry wire cutting, remote operated

- Demolition of other materials by use of:
  - Plasma cutter (steel, aluminium)
  - Dry wire cutting of horizontal beam tubes (concrete, steel, aluminium, lead)
  - Saw, handheld (aluminium tank)
  - Flame cutter (steel, pipes)

The following pictures (Figs. 21 – 26), show the progress of the demolition of the concrete shield. The work carried out by the external contractor was conducted without any significant problems and on time.

Fig. 21 Start of demolition from platform.

Fig. 22 Chimney down, reactor tank exposed.

Fig. 23 Lower part, steel face plates being removed. Horizontal tubes cut out with wire.

Fig. 24 Radioactive parts exposed.

Fig. 25 Almost there…

Fig. 26 Demolition completed.

Presently, the dismantling and closing of the work site (e.g. scaffolding, etc.) remains, followed by the free release of the building during 2008. In general, it can be concluded that the decommissioning of the DR 2 (though the final measuring and free release of the building remain) has been satisfactory, the work performed in time and as planned.

3 Interactions with CRP members and others

The participation in the CRP and interactions with CRP members outside the RCM meetings held throughout the CRP project has been fruitful, in particular to the DR 2 decommissioning.

The decommissioning of the ASTRA reactor in Austria was performed prior to – or ahead of – the DR 2 project. Being of a similar type as the Danish reactor, extensive contact and discussions has been made between the CRP members representing the two projects. A particular focus has been on the method for demolition of concrete structures and on the measuring of possible Wigner energy in the graphite stringers.

4 Conclusions

The various tools presented here and the test and practical use of them, the advantages and disadvantages, the limitations and so forth, is by DD regarded as highly valuable experience. The experience and findings during the projects are all recorded. Particular findings and observations are officially reported. All knowledge and know-how gained during the DR 1 and DR 2 projects are already incorporated into the next upcoming projects (e.g. decommissioning of the Hot Cells facility).

It is generally concluded that the projects have been conducted according to plan and the results have been satisfactory. From the point of view of DD it is found that DD itself was eventually the 'bottle neck' in various operations. This was found, e.g., for the established Waste Documentation System (WDS) that was only fully implemented during autumn 2006. For the DR 2 project this resulted in the assignment of one DD-staff almost full time for this purpose. Presently, DD is confident in the use of the WDS system and the software is running properly. The procedures for control measuring of waste and equipment changed due to the implementation of a certified quality assurance system for the free release of materials and equipment during spring 2007. This was a requirement from the authorities and created some turbulence and misunderstandings in the beginning. It has also been found that DD can improve its processes of decision making on, e.g. health protection, working methods, safety requirements and the appointing of type of containers to be used for various materials. In short, the organisation and management of project is a continuous process of improvement.

The participation in the CRP work and the interaction with CRP members have been useful to the decommissioning projects. It has been purposeful to discuss various tools, methods and other technical issues with the experts and highly experienced members of the CRP. In addition hereto, the CRP process and regular RCM meetings, require a somewhat unusual — i.e. different from standard DD process — evaluation and revision of the decommission work and progress. This 'CRP approach' can be seen as a valuable contribution to the typical planning and management process in decommissioning.

ACKNOWLEDGEMENTS

The support and inspiration from the participants of the CRP and — not least — the IAEA's scientific secretary, M. Laraia, is much appreciated.

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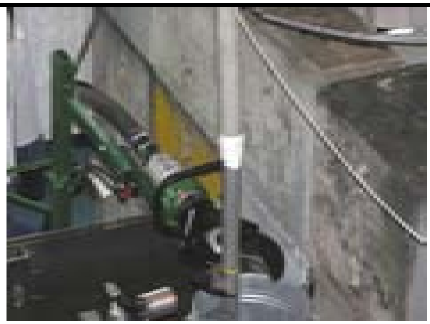
Table 3. Tools considerations

Tool	Picture	Price [€]	Advantages	Disadvantages	Worth buying?
Hydraulic cutter		6.400	Fast cutting Remotely operated “Handy”	None really.	Was it necessary for DR 2?: Yes Has it been as useful as expected?: Yes Can it be used in other projects?: Yes Was it worth buying?: Yes
Automatic band saw		5.300	Fast cutting Limited need for manual operations Easy to move	Requires water (contamination risk). Limited to hard/solid material (aluminium, steel, concrete) – not useful for resin with steel balls!	Was it necessary for DR 2?: Yes Has it been as useful as expected?: Not quite Can it be used in other projects?: Yes Was it worth buying?: Yes
Hydraulic press		12.700	Reduces the waste volume Easy to move	Radioactive particles may be pressed into the plates of the press (difficult to clean). Suitable mainly for aluminium components/ soft materials	Was it necessary for DR 2?: (Yes) Has it been as useful as expected?: Hardly Can it be used in other projects?: Yes Was it worth buying?: Yes/No (Should have been more powerful)

VRDOSE™ AND EMERGING 3D SOFTWARE SOLUTIONS TO SUPPORT DECOMMISSIONING ACTIVITIES

Experiences and expectations from development and deployment of innovative technology

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Abstract
"Innovative technologies for decommissioning", the subject of the Coordinated Research Project (CRP), is a vast field, ranging from the latest water jets and wire cutters, to gamma cameras and spectrometers or abrasive decontamination. In this paper a small but steadily growing range of tools are discussed, namely those based on Information and Communication Technologies, or ICTs. Our focus will be on 3D solutions, and in particular we describe the experience of Norwegian Institute of Energy Technology, IFE, in development and deployment of the VRdose family of software. However, some of the lessons we have learned are probably also useful for other ICT solutions in this field. Virtual Reality (VR) refers to a group of technologies for visualizing, navigating and interacting in real-time with an environment described by a three dimensional computer model. In some industries, like the space industries, airplane navigation and defence, tools of this genre are already well embedded, specially with focus on training and educational applications. In others, like the nuclear field, they have only recently started to emerge from research projects into everyday tools.

1. Introduction

Working with ICTs, the road between science fiction and old news is a short one. When IFE together with JNC (Japan Nuclear Cycle Institute) embarked on developing virtual reality (VR) tools for decommissioning planning in 1999, this was a highly novel approach. Today, 3D systems utilising real-time rendering are common also in the decommissioning industry, but pushing advanced and rapidly developing 3D technology to project teams that are dealing with a 20 to 100 year-long dismantling project is still quite a challenge.

One of the questions we were invited to ask ourselves during our work in the CRP was "What was the rationale for deciding to employ some specific innovative technology in our project?" In our case, as a research institute, we have also asked ourselves "What ought to be the rationale for deciding to employ our specific type of innovative technology in a project?" Therefore, in the following, though focusing on describing technology solutions and our path towards them in words and pictures, we would also like to venture some thoughts on the latter question.

At IFE, there have been ongoing activities on research and development of VR technology to support industrial needs since 1996. Applications range from design of control rooms and operation centres through training applications like the Leningrad refuelling machine simulator that we will decribe in the next section, to planning for outage or decommissioning. By developing modular software based on open standards and formats, it has been possible to obtain the high flexibility which is needed both for re-deployment and for adaptation to the specific situation at hand. In the decommissioning applications it is, therefore, possible to benefit not only from experiences made in other industries, such as air traffic control or integrated operations of oil fields. It is also possible to re-deploy software solutions.

2. Virtual Reality Tools for training, planning and communication

In the 1990s, building large immersive visualisation systems controlled by expensive high-end computers was still in fashion. A few users at a time would enter a Cave™ or similar system with the 3D model displayed at up to four walls, and sometimes also ceiling and floor. This way of seeing computerized information is, of course, impressive. But does one really need to be immersed so totally into the 3D world to be able to learn the right procedure or make the right decission? And how much shimmering water and swaying branches does it really take to generate the required Sense Of Presence (SOP) for a specific industrial or educational application? Several studies have been performed on this, also at IFE [1], and the conclusion seems to be that interactive 3D visualisation is an efficient mean for cognition improvement when deployed on the right kind of problem, but in most cases, a desktop solution with limited use of "eye candy" is sufficient to achieve the wanted effect. And it is certainly more cost efficient.

The two projects that we will describe in the following are both desktop based. They are however scalable, and in situations where a stereoscopic rendering will give added value, this is possible through simply running the same software on high-end computers with more advanced graphics cards. The forerunners of these projects were started before this CRP, but as some of our input to the CRP has been lessons learned from previous research and how we try to reuse and refine based on these lessons, we find that some of the now "historical" information is called-for also in this report.

2.1. VRdose and the Fugen Decommissioning

2.1.1. The background and purpose of the VRdose project

When it was decided that the Fugen Nuclear Power Station (NPS) was to be decommissioned, JNC, (now merged with JAERI to become JAEA), decided to go for a positive approach to the new situation. One of their aims was to build up a set of methods, competence and knowledge to be available in future decommissioning projects in Japan as well as internationally. An important factor when selecting the ICT solutions for this project was JNC's established aim to turn the decommissioning decision into a fruitful and motivating one. Much of the staff would be kept at Fugen and retrained for the dismantling work. Others were to move to operate other facilities in the same area. For all stakeholders, including staff, keeping motivation up was important. It was thus decided to use, what in 1999, was cutting edge technology, as a part of the decommissioning.

Opportunity was also a major factor. Through many years of research collaboration, JNC and IFE had built up a good basis for new technological development. There were two possibilities for a high tech project. An activity could be established within the Halden Reactor Project, where JNC was a member, or a separate dedicated project could be started under a bilateral contract. The latter was found to be the most desirable, since the software then could be tailored to the Fugen decommissioning needs. As research funding was made available, this became the choice.

The VRdose project started at IFE in 1999 [2] [3], with the aim of improving work planning and communication and reducing individual work doses by using virtual reality techniques. One of the ideas was that being able to view radiation conditions inside a representative computerised reproduction of a work environment, one might increase the understanding of radiological conditions and their impact on health, safety and environment (HSE). This would facilitate the contribution to work plans for both operators and executive decision makers. Another important issue was the previous experience in Halden regarding control room design; using VR tended to increase the involvement and understanding of work situation at all levels of the organisation. This could help increase, both work quality, and work efficiency.

2.1.2. VRdose functionality

The purpose of the VRdose software is to give a quick and unambiguous overview over the known radiation conditions in an environment by visualising radiation inside a realistic 3D model based on geometrical data from the work area combined with measured or calculated radiation values. In Fig. 1 and Fig .2 we see examples of visualised radiation conditions at Fugen NPS and the Halden Boiling Water Reactor (HBWR) respectively. Colour scale and amplitude to indicate radiation levels are configurable by the user.

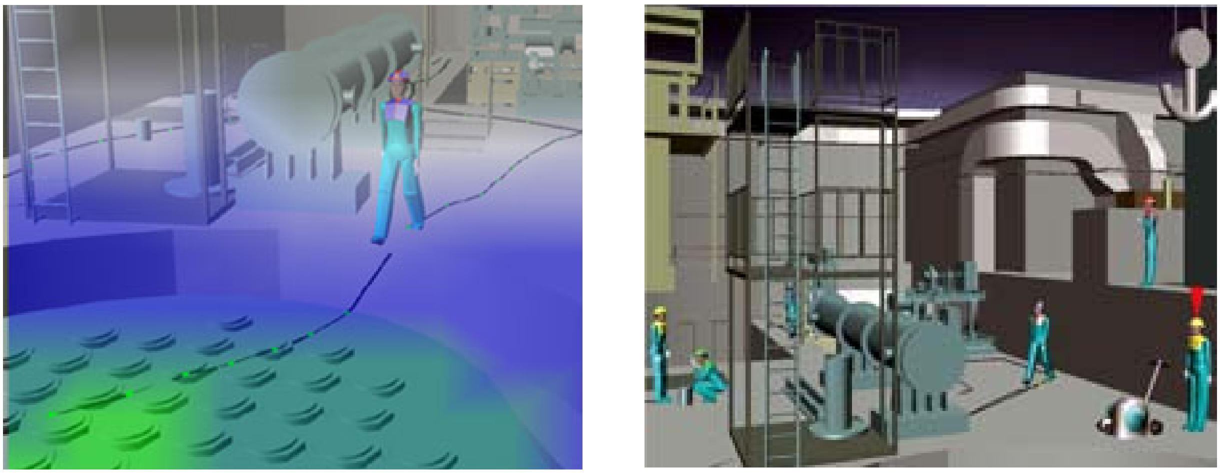

Fig. 1. Radiation and work visualization at Fugen NPP.

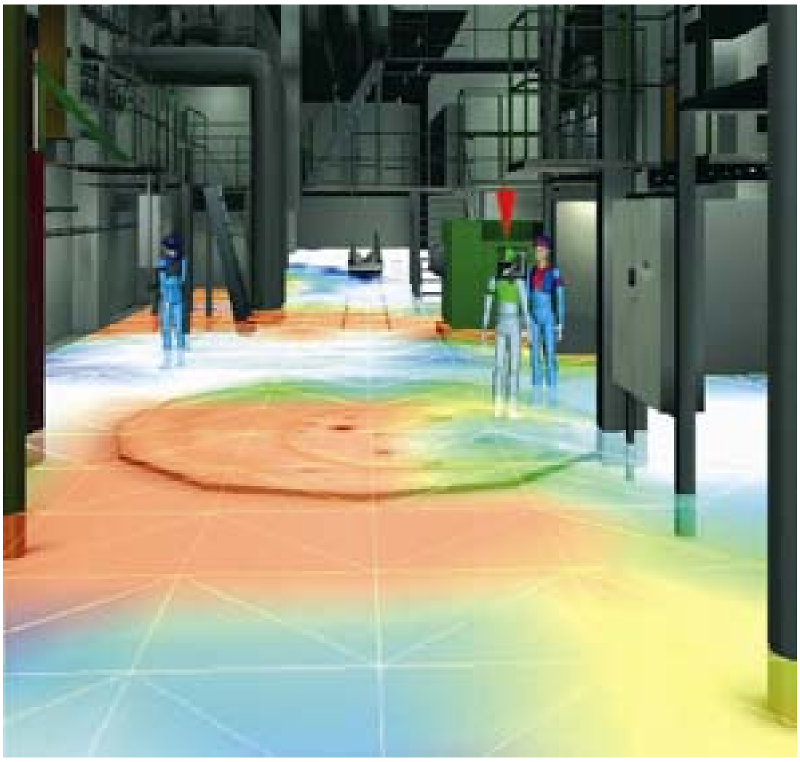

Fig .2. Test work scenario in the Halden Reactor.

Inside these 3D environments, detailed work scenarios are recorded as a kind of editable 3D video, and estimated radiation doses can be obtained for all participants in the virtual work team. Fig. 3 shows an example of this. Such scenarios can be stored and edited, and one can optimize a work plan so that periods spent in high level radiation areas are minimized. The software can also differentiate between gamma radiation and intake doses if alpha and beta radiation data is available. Depending on the type of radiation, the user may also select between linear interpolation or source estimation.

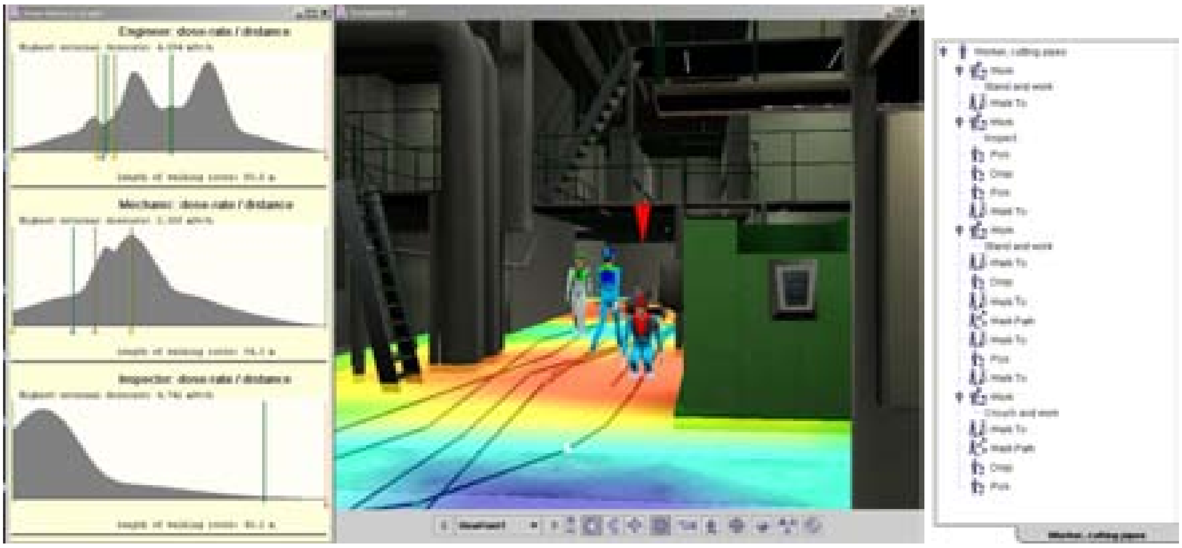

Fig. 3. Dose graphs describing the estimated radiologigal effect of a chosen work procedure.

The typical end user of this software is a member of radiation protection staff involved in dismantling or training activities and management planning or demonstrating work procedures. The staff gets a better overview and with a heightened shared situation awareness [4] teams are likely to make better decisions. An example is that having a three dimensional picture of radiation situation in your mind you have a better posibility of minimizing your own doses. Through recording work scenarios and evaluating doses in advance for all work team members, it is also easier to identify operations that are likely to yield a high dose cost, and in many cases, plans can be adjusted to reduce doses.

Another application of this software is for communications with stakeholders. Local authorities, media and school classes visited the VR facility at Fugen to learn about the decommissioning plans. Here the 3D model, work plans and radiation levels are shown on a large screen with the possibility for passive a stereo view if wanted. The same facility can be used by groups of staff for discussion and planning.

Although more recent versions have been made of this software at IFE, the Fugen version of VRdose is the most extensive one. In this case, all inventory in the NPS has been entered into a database. By clicking on an object in the virtual environment available information on the real object can be accessed. Database information on real life operators is also available and can be compared to the virtual doses.

2.1.3. Testing and deploying VRdose at Fugen

At present VRdose is a part of the advanced technology used at Fugen [5][6]. It is a tool for visualising work situations and radiation conditions inside a Virtual Environment (VE) of a workplace. The VE can be obtained in various ways, as described in [7], and there are also several options for creating the radiation map. After the VRdose project was completed in March 2004, the system was used at the Fugen NPS. The last update of VRdose was delivered to Fugen in 2006. The following shows an example on how VRdose is used.

A floor drain pipe needed replacement in the building where the radioactive waste at Fugen NPS is treated. A study was made in connection to this work operation in order to compare the real dose that the workers were exposed to and the dose predicted by VRdose. The work period was from July 2005 to September 2005, and results from this study were kindly made available to HVRC by JAEA in 2006. The screenshot to the left in *Fig. 4* shows the VR model of the work area with 3D radiation visualisation. At the bottom right, we see VR operators entering the work area, wearing intake protection gear. In the upper right corner, the workers movements are indicated as black paths on a colour coded 2D radiation map.

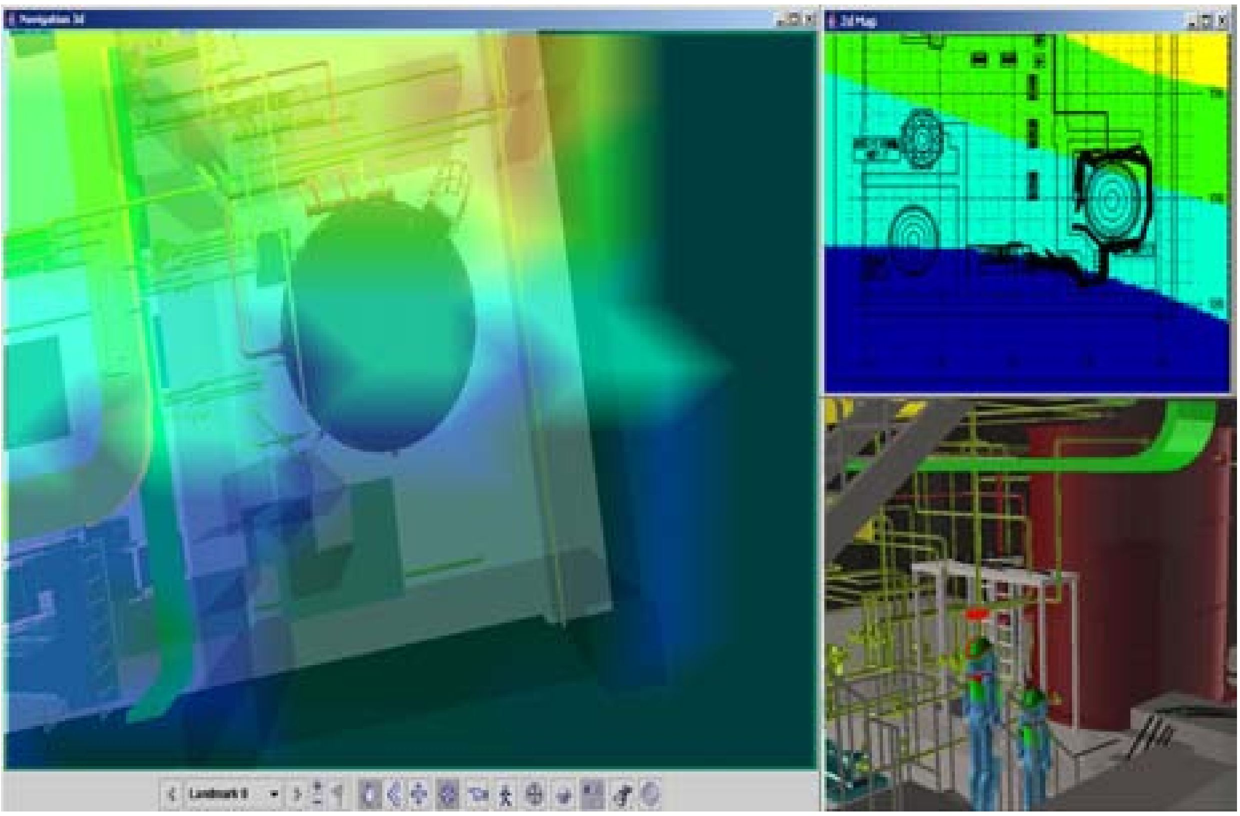

Fig. 4. VRdose at Fugen: Planning the replacement of a floor drain pipe.

After the work was completed, the results from VRdose simulations were compared to the real life experience, as shown in *Fig. 5*. The conclusions of the Fugen staff were that the VRdose predictions tended to be conservative, as also demonstrated in previous tests [2] and that the average ratio estimated dose/real dose was approximately 1.4.

The worker doing the cutting of the pipes has received a slightly higher dose than VRdose has predicted. One possible reason for this is the way VRdose is estimating intake doses. Gamma radiation, which was the main concern during the first phased of project work, is treated very thoroughly by the VRdose and Hitachi DRES software , while only simple linear interpolation is used for the measured intake values. However, dismantling work often increases the airborne contamination level which leads to increased intake doses. Adding a proper calculation of this effect is therefore next on the VRdose work list, and improved algorithms will be implemented.

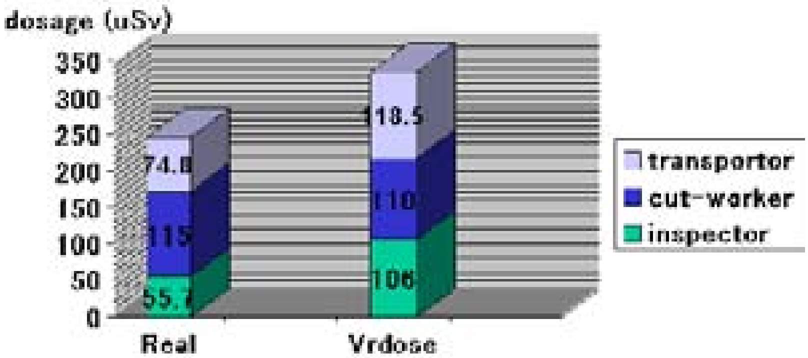

Fig. 5. Fugen analysis of VRdose results.

It is not worrying that the dose of the inspector calculated by VRdose is close to double the real dose. VRdose should give a conservative estimate when used correctly, and as all worker movements are not added to a work scenario, the virtual workers are often recorded as standing still in high radiation areas, while the real worker, when aware of the radiation situation, will keep stepping back between operations.

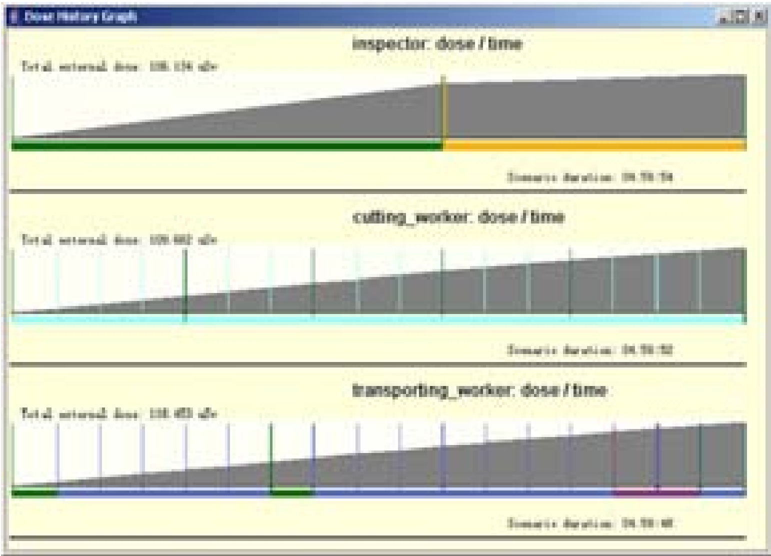

Fig. 6. VRdose estimates of radiation exposure.

If we look at the dose graphs in Fig. 6, we can see from the colour coding under each graph what each virtual person was doing when the dose was received. Green means walking, yellow means working or stopping up to inspect something. We can see that the virtual inspector is moving only at the beginning of the scenario. He is then recorded as standing still in one spot, close enough to the work to see what he needs for the last 2 hours of the scenario. The real life inspector was probably moving around more, and only went as close as the assumed position when it was necessary.

2.2. *VRdose descendants*

VRdose modules have later been re-applied. First in a project for ENEA (Italy), called VirtualDecom, which started up in 2002 but had its final update delivery in 2006. Here, the purpose of work simulations and radiation visualisation was to visualise newly planned procedures for dismantling of plutonium contaminated glove-boxes in a designated dismantling facility [8]. An interesting lesson learned from this project was that when procedures are fully visualised in 3D, with people and all equipment and tools involved, their implications are likely to be better understood. In this case, the procedures to be visualised were changed several times, also due to reasons other than radiation levels. In some cases there was too little space for the dismantling task etc. For example, a curtain, that needed to be closed, might be too tight because of the size of the planned forklift. The impression we received was that work tasks become more tangible for those who perform the procedures when visualised this way.

In 2005 another application was developed based on VRdose philosophy in a research project for the Japanese power company TEPCO, working together with their research institute TEPSYS and the hardware company Chiyoda Technol. This application, called Virtual Live Dosimetry (ViLDe), visualises online radiation measurements in nuclear environments, and is now being introduced also to a completely different industry, namely hospitals.

VRdose has been used in several human-factors experiments for the OECD NEA Halden Reactor Project [9], yielding not only experimental results on radiation awareness but also valuable feedback to the developers on usability issues and desired functionality [10] [11]. A close cooperation between human factors specialists, computer scientists and radiation protection experts has been necessary and fruitful. As a part of the software developed specially for Halden Project members (for use in experimental and ICT research), modules with functionality much like VRdose are also being developed. This software is called the Halden Viewer [12]. In the 2009-2011 Halden Reactor Research Programme, decommissioning will be a separate activity.

2.3. The Chernobyl Decommissioning and reapplication of the Leningrad Refuelling Simulator

2.3.1. The Chernobyl decommissioning – background and status

The well known explosion hit at the RBMK No. 4 of the Chernobyl Nuclear Power Plant (ChNPP) in April 1986. It destroyed the reactor and spread radioactive material into the surroundings. With this background, the entire plant was closed in December 2000. The decommissioning of units No. 1, 2 and 3 at the plant is now being prepared. The overall time schedule for the decommissioning indicates that the final shutdown and preservation will take about 10 years. The safe enclosure is set to last 100 years followed by the final dismantling, taking 8–10 years.

At present, the activities are concentrating on the final shutdown. The final shutdown and preservation stages mean that the NPP units must be put in a condition excluding possible future operation in addition to providing safe storage of radioactive material and exposure sources for a defined period of time.

The dismantling of the reactor building will involve several stages. The first one is dismantling the refuelling machine followed by lowering of the main crane down on to the RM crane bridge for temporary storage. The removing of the reactor hall roof and upper outer walls will then be the next task. A new roof will then be put in place, and a new crane placed at the lower outer walls. The auxiliary systems and rooms are the next to be removed before the final dismantling of the reactor core and the decommissioning of the remaining reactor building. The state of the final plant site, when all systems are removed from the buildings and other radiological cleanup has been performed, is today foreseen to be a brown field release. This means that radiation levels will be low enough for restricted use, like industrial purposes.

2.3.2. Introducing Virtual Reality tools from the Leningrad Nuclear Power Plant at ChNPP

The decommissioning of a RBMK reactor is no less complicated than running it. When planning and performing the dismantling, the need for information about the actual design and the current state of the plant is of highest importance for the staff, the authorities and the public in order to perform the work safely, efficiently and with acceptable economic costs.

As part of its safety program for nuclear power plants in Russia, Central and Eastern Europe, the Norwegian authorities have, since 1999, funded a project at the Leningrad Nuclear Power Plant (LNPP) in Russia. The reactor type at the operating LNPP is RBMK, similar to the one found at ChNPP. It is designed for online refuelling during full power using a special refuelling machine (RM). The goal of the project at LNPP has been to introduce VR as an important pedagogical means for better training operation and maintenance of the refuelling machine. The LNPP project has been a close teamwork between LNPP, IFE and the Russian Research Centre “Kurchatov Institute (RRC KI), where IFE has been the project manager and RRC KI a subcontractor to IFE.

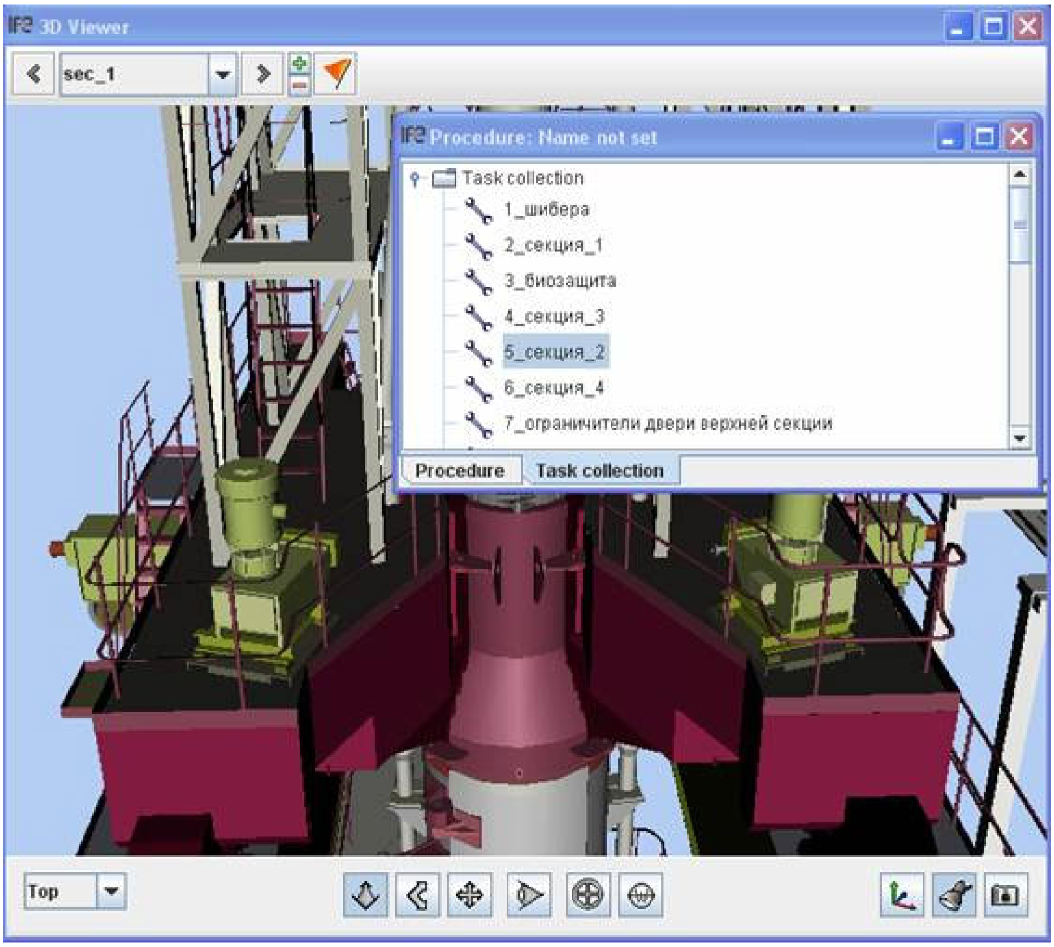

Fig. 7. Procedure creation in VR.

VR is an effective way of visualising, interacting with and navigating through an environment described by a 3D computer model. The VR technology is useful for planning, training and presentation because it offers a realistic way of visualising the real world in 3D together with the potential for direct user interaction and system feedback. It is particularly suited for situations where the spatial skills are important. One can interact with 3D data in a virtual environment, providing a potentially powerful interface to both static and dynamic information.

The results from the project at LNPP are two closely-related VR applications: A simulator for training the refuelling operation and a procedure training system for training procedures related to the maintenance of the safety critical interior parts of the RM. As part of the project at LNPP, a large number of detailed 3D models of the RM and reactor hall have been developed.

The success of introducing VR at LNPP was demonstrated at LNPP to staff from ChNPP in 2005. The ChNPP management saw this existing technology as an opportunity and decided that VR may play an important role as a tool for visualising the dismantling tasks. Together with IFE, the parties applied to the Norwegian Ministry of Foreign Affaires for a pilot project using VR as a tool in the decommissioning.

The pilot project was approved by the authorities and started in late 2006 with completion by the end of 2007. The long-term plan is that the Chernobyl Decommissioning Visualisation Centre (CDVC) will be a VR tool for nuclear decommissioning enabling a group of engineers and others involved in the planning to test out plans and procedures for various work tasks involved in dismantling the nuclear facility. The visualisation centre is also expected to be a valuable tool for making the needed documentation and for training for the dismantling phase. Intensive training before the real dismantling process may be effective in reducing radiation exposure dose, workload and for enhancing safety. In addition, the centre will provide the decommissioning project team with an effective medium for presentations to the public as well as for communicating with management and the licensing authorities.

2.3.3. The Chernobyl Decommissioning Visualisation Centre

Decommissioning safely the intact RBMK reactors at ChNPP will be a long-term technological and organisational challenge because of the complexity of the plant, the need for transferring knowledge to the next generations and the contamination.

Dismantling a NPP requires highly detailed documentation. Unfortunately a large quantity of the documentation at ChNPP has been lost and must be reconstructed. There is also a need to re-establish skills among the personnel, as the equipment has been out of use for a long time. This means that new personnel are unfamiliar with it, and that in the meantime, much of the experienced staff has left the plant. Careful analysis and planning of the dismantling tasks will be important in addition to training the personnel through both, theoretical exercises, and practical rehearsals. The IFE VR technology has shown to be useful in a number of projects for planning, training and presenting the work tasks to be carried out in hazard areas with radiation. It is expected that use of VR technology in the planning will prove to be beneficial both with regards to minimizing the workers' radiation exposure as well as in helping to achieve an efficient use of manpower.

The VR technology is based on the VRdose™ application together with the results from the LNPP project. The participants in the project are ChNPP, IFE, LNPP and RRC KI. The end-user is ChNPP, and the experts from the ChNPP Training Centre, the Decommissioning Division and the Repair Planning Division participate in the work. IFE is the technical project leader and the developer of the software for the Visualisation Centre. LNPP takes part as a technical advisor in the RBMK reactor and RM technology. RRC KI is subcontractor to IFE in the creation of new 3D-dimensional models and in the adaptation of existing ones from LNPP to ChNPP needs.

The first scenario to be implemented will be the dismantling of the RM. Approximately 70% of the 3D models for RM can be obtained from the VR projects at LNPP. The last 30% of the 3D models is being made in cooperation between ChNPP's specialists and RRC KI. Thereby the project saves both funds and time.

Knowledge in the area of VR technology will be transferred from IFE to ChNPP's specialists through the teamwork between the parties, so that after the realisation of the project, ChNPP can implement new decommissioning scenarios in CDVC. This transfer of knowledge includes training in 3D modelling and optimisation for the use of the VR software.

The CDVC offers stereoscopic visualisation of 3D virtual environments, features for dismantling procedure development and documentation in addition to occupational dose calculation. The result from the pilot project is a first version of the Chernobyl Decommissioning Procedure Creator (ChNPP ProCre) developed by IFE with basic functionality for making dismantling procedures.

The procedure for a dismantling task can be put together in ChNPP ProCre by deciding the order in which the items are going to be removed. The first version of ProCre has limited functionality and the calculation of the radiation dose received when following the procedure is not implemented. The documentation developed will be usable for training the personnel before doing the tasks, as documentation for use in the field for doing the job, or when presenting the decommissioning project for the public and the authorities. Thereby, the CDVC is expected to support the ChNPP in preserving the decommissioning expertise and knowledge at the plant in a long term perspective.

The hardware for the centre will be purchased in 2008 and the centre will be established in a separate room at the ChNPP Training Centre in Slavutich. It will consist of a powerful PC with the VR software developed by IFE in addition to two projectors capable of showing stereo and a large wall screen.

IFE and ChNPP will apply to the Norwegian authorities for a continuation of the project in 2008-2010 to further enhance the CDVC and ChNPP ProCre to meet the requirements of ChNPP staff based on feedback from experience using the current facility. The Norwegian government finds it important to support projects with the purpose of increasing safety at nuclear facilities in countries close to Norway. The complex task of decommissioning the ChNPP is regarded by the government as such a project.

3. Discussion of future and ongoing work

For the use of VRdose at Fugen, the Hitachi DRES system [13] that calculates dose-rates based on data on contamination and activity of components in the inventory data base are designed to be compatible and to handle the same (simulated) data. In other applications it is an identified weakness that the collection of the radiation data and the visualisation of available data are not always handled by the same technology. For a field manager planning a dismantling operation, the usability of a tool like this would increase significantly if he/she were able to collect, study and visualise all the radiological data assisted by the same software framework. And for a decommissioning project manager, being able to evaluate and aquire technology (hardware and software) to perform these activities "all-in-one" would profoundly simplify matters. Another important factor is the kind of data available. Though VRdose can be used to visualise any volumetric data, the main application so far has been to visualise estimated dose-rate.

The French company Electricité de France (EDF) has been the first nuclear energy producer in the world to equip all its NPP radioprotection personnel (58 operating reactors) with specially-designed, portable, CdZnTe gamma spectrometers and associated spectral analysis software. Their aim in this is to identify the radioactive isotopes (e.g. corrosion products) responsible for operator doses so that maintenance operations may be optimised. The experience they gain in this would clearly be useful also for the decommissioning community.

IFE and the (EDF) have now started collaboration on combining the EDF and IFE experience and technology into tools and methods to make a work team able to retrieve and visualise not only dose-rate, but also sources and identified isotopes. The result will be an extended VRdose type of application with direct linkage to tested off-the-shelf hardware that may be combined with it to make a "suite". By being able to see not only one or a series of dose-rate situations, but rather the exact sources of the situation and their properties, better planning for shielding and dismantling can be achieved through raised radiation awareness. In this way, significant dose reduction should be possible.

4. Interactions with CRP members and others

We have had very useful discussions with Yuri Lobach, Ukraine, in the start up phase of our Chernobyl activities. Silvio Fabbri, Argentina, has received a test version of VRdose software to try out at their facilities. There have also been discussions on future collaboration with Vladimir Daniska, Slovakia and Sergey Micheykin, Russia. And we have kindly received characterization data and descriptions from Kurt Lauridsen, Denmark to use in our research.

5. Conclusions

It is our experience that advanced interactive 3D visualisation technology can be helpful in work planning and training and as a mean for giving a quick overview of radiological conditions to a work team. As radiation conditions and work procedures are typically changing with the transition from operation to decommissioning, it also seems likely that benefits from such tools will increase in a decommissioning setting. At present, 3D solutions are no longer theoretical, and advanced 3D tools are in one way or another deployed in most decommissioning projects.

Virtual reality tools like VRdose and CDVC, and other video game-like applications may give added value also as so called “wow”-factors, aiding in public relations and recruitment. It is however important to keep a clear focus, and to acquire or develop such tools based on real life requirements and real life problems, as a shimmering demonstrator with no real purpose quickly looses the work team’s interest and saves neither money, time or dose.

We have benefitted significantly from taking part in this CRP and from being able to discuss technology potentials with other players facing decommissioning challenges. Hopefully, this is also helping us keep our man-technology-organisation research focus aligned to real life decommissioning challenges.

ACKNOWLEDGEMENTS

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TECHNOLOGY DEVELOPMENT ON THE DECONTAMINATION AND DECOMMISSIONING OF THE NUCLEAR RESEARCH FACILITIES IN KOREA

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Abstract

This paper covers some of the above R&D results performed from 2004 to 2007 at KAERI in the field of dry decontamination, decommissioning and decommissioning waste treatment. The scope of the R&D is divided into three parts. For dry decontamination technology development applicable to a dry hot cell maintenance such as DUPIC (Direct Use of PWR spent fuel in CANDU reactor) demonstration hot cells, three dry decontamination technologies such as a CO_2 blasting, PFC (PerFluoroCarbon) soaking and plasma decontaminations were selected depending upon the contamination characteristics such as the internal or external contamination and the fixed or non-fixed contamination levels. The second part of the R&D is the decommissioning technology development applicable to KAERI's on-going decommissioning project. The R&D results about the in-situ radioactivity measuring equipment applicable to contaminated pipe internals, a digital mock-up system for the dismantlement of nuclear facilities, and the technology for the treatment and management of neutron irradiated graphite are presented. For the decommissioning waste, the melting decontamination technology has been studied to recycle the radioactive metallic wastes generated from dismantling KAERI's two research reactors (KRR-1&2) and a uranium conversion plant (UCP).

1. Introduction

Decommissioning of two research reactors at KAERI started in 1997 and will be completed by 2008. One research reactor was completely dismantled by 2005 and another reactor is waiting for a final decision regarding option: completely dismantling it or its conversion to a museum.

Decommissioning of a uranium conversion facility at KAERI started in 2000 and will be completed by 2009. Most parts of the facility have been dismantled and decommissioning waste management, including site restoration, is under way.

The decommissioning wastes generated up to 2007 are listed in the Table 1 and Table 2.

Table 1. Decommissioning waste from KRR-1 and 2 [tonnes]

	radioactive	for release	total
metal	18	163	181
concrete	260	1,746	2,006
others	17	35	52
total	295	1,944	2,239

Table 2. Decommissioning waste from a uranium conversion plant

<table>
<tr><td colspan="2">Waste</td><td colspan="2">Amount, tonnes</td></tr>
<tr><td rowspan="2">Metal</td><td>Carbon steel</td><td rowspan="2">179</td><td>106</td></tr>
<tr><td>Stainless steel</td><td>73</td></tr>
<tr><td colspan="2">Concrete</td><td colspan="2">11</td></tr>
<tr><td colspan="2">Cable</td><td colspan="2">6</td></tr>
<tr><td colspan="2">Uranium</td><td colspan="2">3</td></tr>
<tr><td colspan="2">Others</td><td colspan="2">13</td></tr>
<tr><td colspan="2">Liquid Waste</td><td colspan="2">14</td></tr>
<tr><td colspan="2">Total</td><td colspan="2">226</td></tr>
</table>

KAERI has many hot laboratories such as a Post Irradiation Examination Facility (PIEF) and an Irradiated Material Examination Facility (IMEF). Most of them are old, so a large scale maintenance and refurbishment of them is expected in the near future.

KAERI has carried out D&D R&D funded by government to support the on-going decommissioning project and a future refurbishment of the hot laboratories at KAERI.

The first stage of these R&D programs was completed from 2001 to 2004. The second stage of the R&D programs was finished from 2001 to 2004 and some of the results were reported to the IAEA CRP.

2. D&D and its waste treatment technology development

2.1. Dry Decontamination technologies applicable to dry hot cells

Three dry decontamination technologies were developed for the maintenance and refurbishment of the dry hot laboratories such as the PIEF (Post Irradiation Examination Facility), IMEF (Irradiated Material Examination Facility) and DUPIC hot demonstration hot cells at KAERI. For the loosely adhered contaminants, carbon dioxide pellet blasting and perflurocarbon (PFC) decontamination technologies were selected and for fixed contaminants, plasma decontamination technology was chosen in the studies.

The overall dry decontamination process concept is listed in the Figure 1.

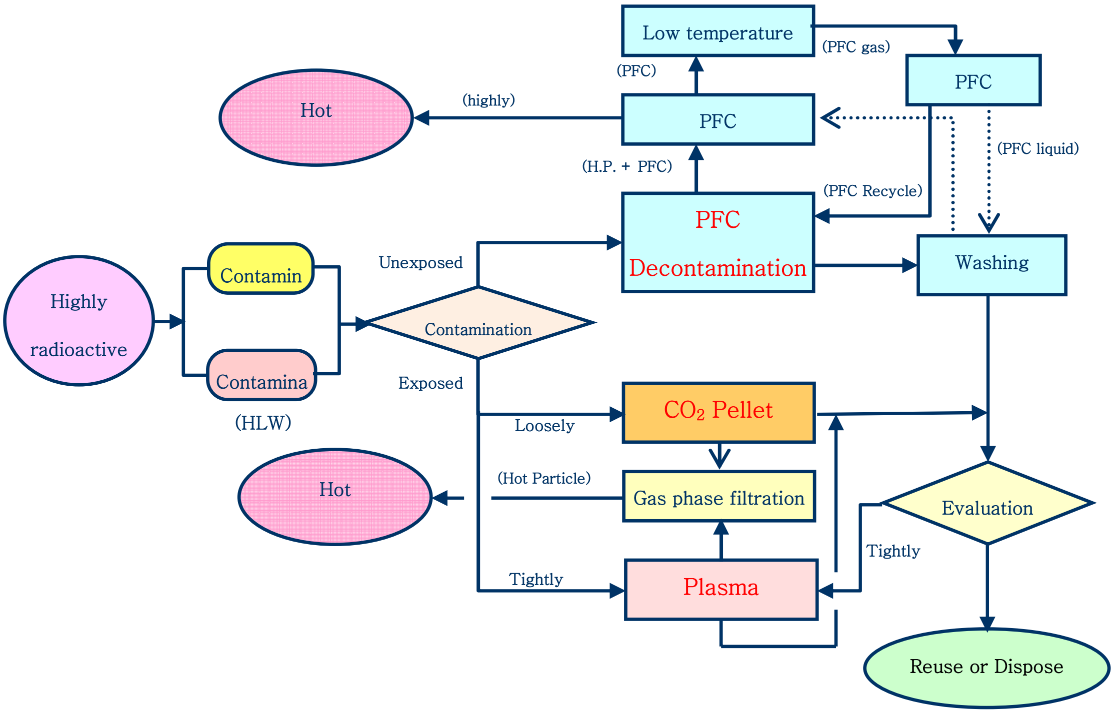


Fig. 1. Conceptual flow diagram of dry decontamination.

2.1.1. CO_2 pellet blasting

A high speed jet of carbon dioxide is forced through a converging-diverging nozzle, which results in the formation of solid CO_2 ice flakes in a diverging part of the nozzle. Dry ice pellets consist of solid CO_2 at a temperature of -78.5°C formed by relieving the CO_2 to an atmospheric pressure of 1 bar at a temperature of -80°C. A hydraulic stamp presses the snow through a mould in a pelletizer and the pelletizer produces cylindrical dry-ice pellets with a diameter of between 1 and 6 mm and a length of 5 to 15 mm.

A CO_2 pellet blasting equipment with a capacity of 0.5 kg/min and 30 kg of dry pellet per batch was designed and fabricated to study the process parameters such as the physical properties of the pellet and application pressure, distance and time. The decontamination equipment consisted of a pellet extruder and a pellet blaster. The blaster has the functions of purging, blasting and particle collection.

Examples of the experimental decontamination results are shown in Figure 2.

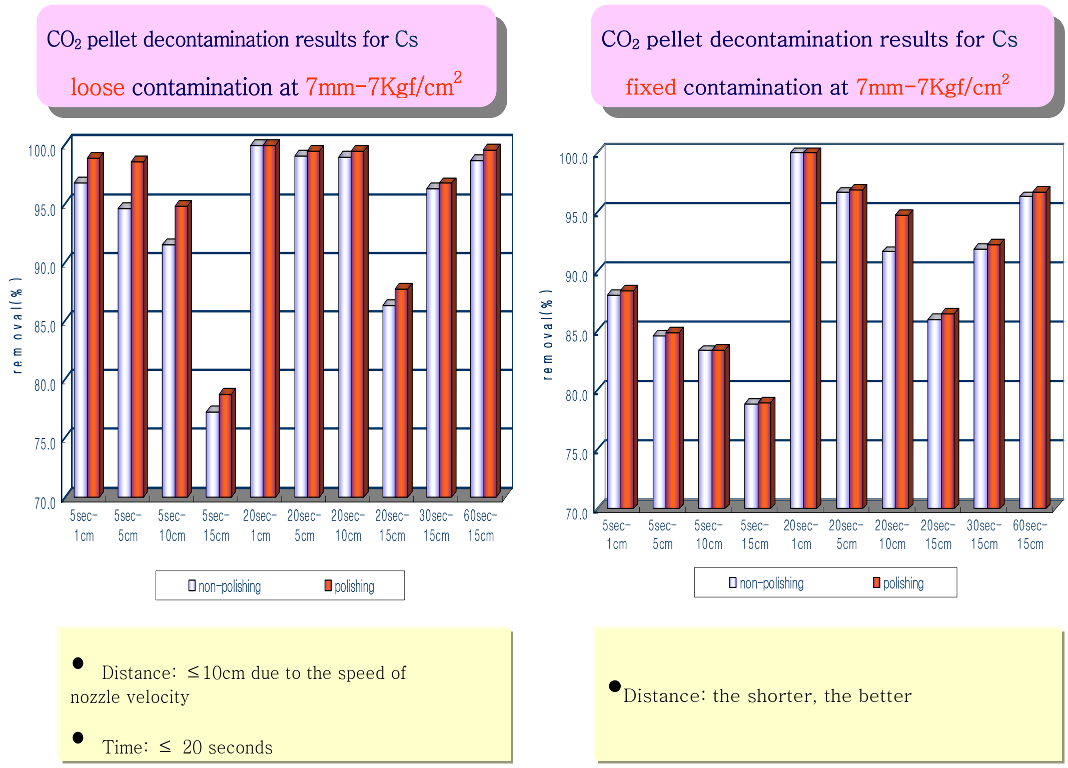


Fig. 2. CO2 pellet blasting decontamination for the contaminated particles.

The decontamination behaviors were tested with the three kinds of the contaminants: particulates, chemical compounds and grease/oils on the surface of the SUS 304 plates. The experimental results showed that the decontamination fraction for the particulate contaminants were about 99% and the larger particles of 5 to 10 um were more easily decontaminated than the smaller particles around 1 um. It was also shown that the decontamination performance for the chemical compounds was about 90% and those for the high viscous grease/oil contaminants needed an application time of more than 120 seconds.

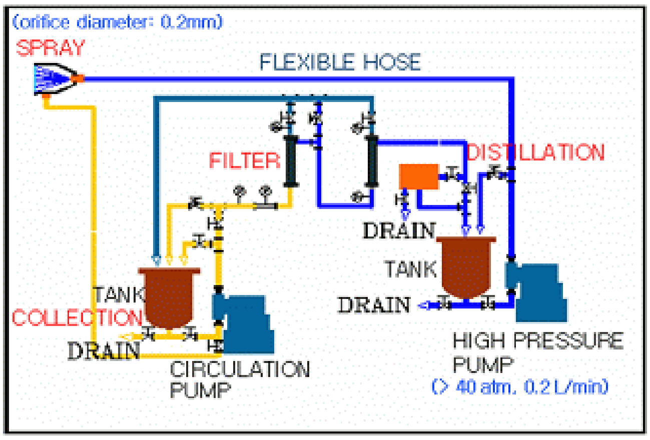
The use of the developed CO_2 pellet blasting technology proved to be very effective in the removal of a fixed particulate contamination as well as loose. During the CO_2 pellet blasting, it is evaluated that a suction unit, which is able to catch the removed contaminants, is needed to prevent a recontamination and to obtain better decontamination performance.

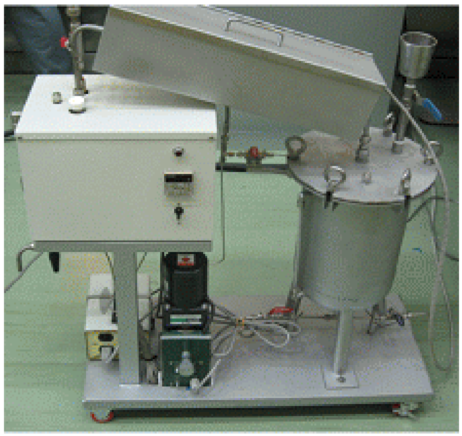
2.1.2. PFC dry decontamination technology

The PFCs are liquids of a high density, colorless and ataxic, characterized by a low refraction index, a low surface tension and a low dielectric constant. Unlike halofluorocarbons, PFCs are not ozone depleting compounds and for this reason they have recently been substituted by halofluorocarbons in several technological applications.

As a part of our project, the PFC ultrasonic decontamination technology development was performed in 2004. A PFC ultrasonic decontamination using the several shapes of metal specimens was performed in a PFC solution. For all the tested specimens, we found that an ultrasonic decontamination was satisfactory. As the PFC solution is a non-conducting substance and easily separated from the contaminants, the PFC decontamination process is a promising method to decontaminate metal surfaces loosely contaminated with radioactive particles.

Based on the ultrasonic decontamination test results, the PFC spray decontamination technology development has been carried out with the feasibility of reuse of the PFC solution by distillation method since 2005. A schematic diagram of the PFC spray decontamination process and the apparatus is shown in Figure 3. The decontamination equipment consists of a decontamination module, a suction module, a filtration module and a distillation module. Each module is connected with a 1/4 inch flexible hose. The spray pressure was 41 kg_f/cm^2, the orifice diameter was 0.2 mm and the spray velocity was 0.2 L/min.





(a) (b)

Fig. 3. Schematic diagram of the PFC spray decontamination process (a) and the apparatus (b).

In order to examine the characteristics of the PFC spray decontamination process on the several shapes of the metal specimens, the decontamination tests were performed in the hot cell of the radioactive waste drum examination facility in KAERI. The radioactivity of the specimens before a decontamination is in the range of 732.4 to 931.5 Bq/1000 cm^2 for ^{60}Co, and it is in the range of 386.2 to 942.2 Bq/1000cm^2 for ^{137}Cs. From the test results, we found that the decontamination factor was in the range of 9.6 to 62.4. When the decontamination efficiency of ^{60}Co was high, then the decontamination efficiency of ^{137}Cs was also high. As the surface roughness of the specimen increases, the PFC spray decontamination efficiency decreases.

In the Sonatol process, a PFC ultrasonic wave is applied to several kinds of materials for one hour. As a result, 99.7 % of the contaminants was removed by the Sonatol process. The decontamination efficiency of the PFC spray decontamination is comparable to the decontamination efficiency of the Sonatol process.

As a result of multiple applications, the PFC solution becomes dirty. The contaminants in the solution are removed by distillation and the solution can be recycled. By distillation, more than 97.5 % of the PFC was recovered. As the PFC solution is a non-conducting substance and easily separated from the contaminants, the PFC spray decontamination process with the recycling of the spent PFC solution by a distillation will be reliable to decontaminate a loosely contaminated area. This process will be applied to the decontamination of the highly radioactive hot cells in KAERI.

2.2. Decommissioning technologies for research reactors

2.2.1. In-pipe radioactivity measurement system

Generally, a large amount of wastes are generated during the decommissioning of nuclear facilities. These wastes are contaminated with various types of nuclides emitting alpha- and beta particles, or gamma rays. The contamination level of the decommissioning wastes must be surveyed for free release or reuse.

The surface contamination of alpha and beta activity needs to be simultaneously measured in the nuclear facilities. Such a contamination measurement could be conducted by a proportional counter or phoswich detector. But the proportional counter is very difficult to make for insertion into a small size pipe. It is possible to manufacture a small size phoswich detector for a simultaneous counting of alpha- and beta, which is widely used in various fields.

In this period, a phoswich detector for simultaneous counting of alpha- and beta activity in a pipe was developed. The scintillator for a counting an alpha particle has been applied to a cylindrical polymer composite sheet with a double layer structure of inorganic scintillator ZnS(Ag) layer adhered onto a polymer sub-layer. The sub-layer in an alpha particle counting sheet made of polysulfone works as a mechanical and optical support. The ZnS(Ag) layer is formed by coating a ternary mixture of ZnS(Ag), paste (polysulfone or cyano resin) as a binder and solvent onto the top of a sub-layer via a screen printing method. The plastic scintillator was simulated by using a Monte Carlo simulation method for detection of the beta radiation emitted from internal surfaces of a small diameter pipe. Simulation results predicted the optimum thickness and geometry of a plastic scintillator at which an energy absorption for a beta radiation was maximized. The characteristics of the detector to be fabricated were also estimated.

A conceptual diagram of the in-pipe monitoring system for a simultaneous counting of alpha-, beta-, and gamma-rays is shown in Figure 4. The detecting part was constructed as a phoswich type for a simultaneous counting of alpha- and beta-rays in a pipe. Each radiation must be discriminated in a phoswich detector to allow for a simultaneous measurement of different radiation types using a single-detector system. A pulse-shaped discrimination and a pulse-height discrimination are generally used to discriminate each radiation type. Pulse heights generated by alpha- and beta were discriminated using the energy level discriminator for the pulse-height discrimination method. But, accurate measurements cannot be performed because of an overlap of the alpha and beta events for a low level activity. Therefore, the pulse-shape discrimination method is generally used to discriminate each radiation. The pulse-shape discrimination method discriminates the rise-time of a scintillation formed in each scintillator. For the detection ability of the phoswich detector, alpha-particle emitting nuclide, ^{241}Am, and beta emitting nuclide, $Sr/^{90}Y$, were used. The scintillations produced by an interaction between radiation and scintillator were measured by a PMT (photomultiplier tube).

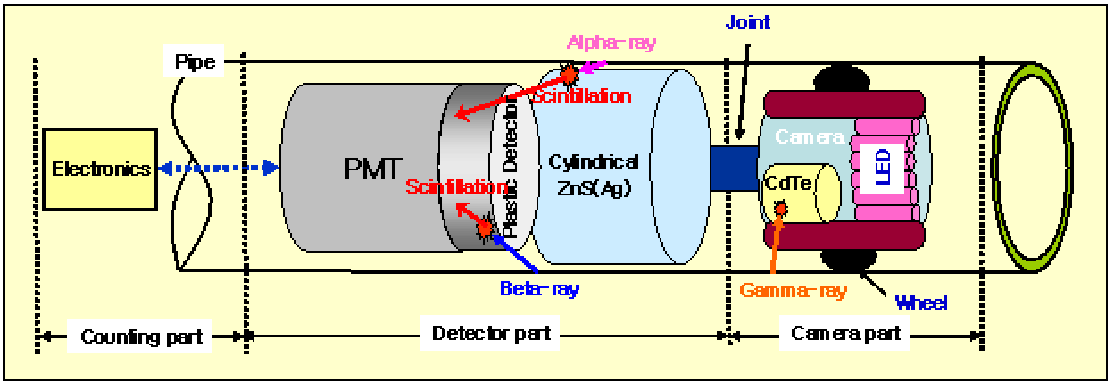


Fig. 4. Conceptual design of the in-pipe radioactive monitoring system.

The prepared cylindrical ZnS(Ag) scintillator had sufficient mechanical strength, optical property, and a good detection ability for alpha-rays. Also, the plastic scintillator for beta detection with an optimal thickness and optimal distance between the detectors and counting position revealed a good detection ability.

In the near future, the developed in-pipe monitoring system will be tested at decommissioning sites such as the KRR-1&2 and uranium conversion facility. And, after comparing the contamination data taken by conventional methods, it will aid in an understanding of any problems with the detector and for improving the monitoring system.

2.2.2. Digital mock-up system for the dismantlement of nuclear facilities

A technology that can reduce dismantling schedule, minimize a worker's dose, and cut down on the decommissioning cost was needed. For this reason, we developed the dismantling mock-up (DMU) system in order to show a dismantling process through animation and to simulate decommissioning information. The DMU system can allow us to experience a dismantling procedure through an animation prior to a dismantling and to optimize the process related to the important parameters. The DMU system is made of several modules such as a 3D CAD modeling, a visualization and assessment of the radioactivity inventory, an animation, a simulation, and an analysis and evaluation as illustrated in Figure 5.

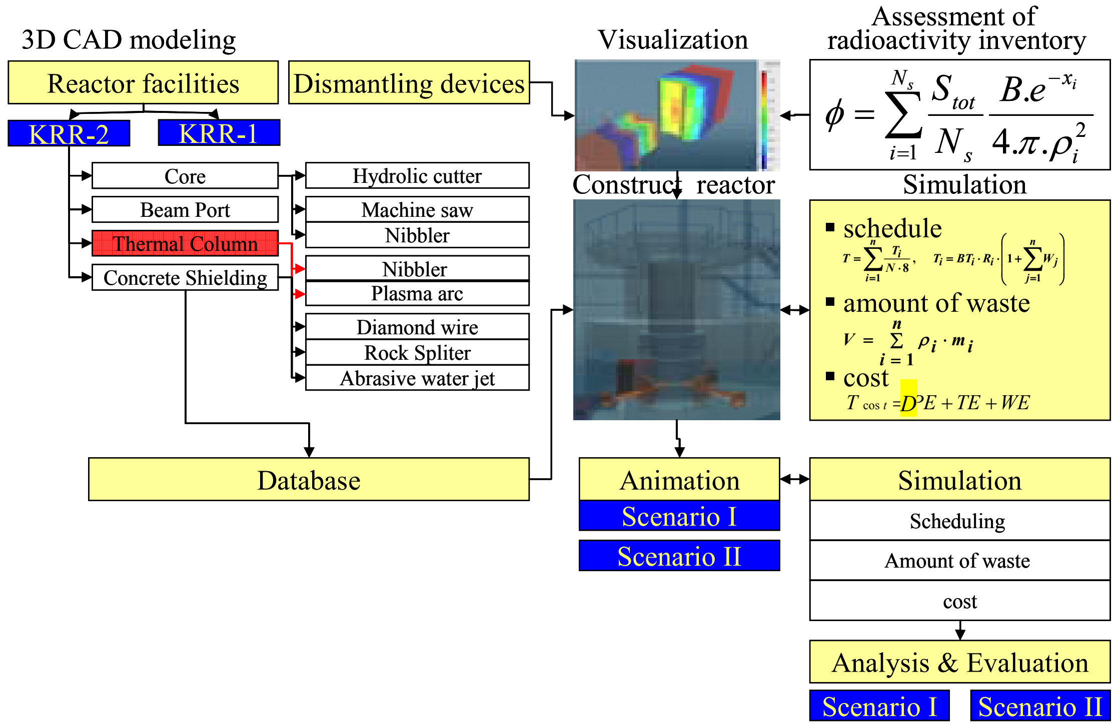


Fig. 5. Schematic diagram of the KAERI's research reactor dismantling digital mock-up system.

Using 3D MAX and AutoCAD, we draw a research reactor facility in order to establish the animation module as follows:

- Research Reactor 1&2
- Thermal Column, Beam Port, and Reactor Core

The DMU system performs the dismantling process through a selected scenario and evaluates the dismantling time and cost according to the dismantling process in the animation module. The animation scenario enables a worker to easily understand a dismantling procedure as it can be viewed from different angles for the major components that compose a reactor. We can view the inner or outer part of the reactor through a walkthrough and transparency evaluation.

In order to evaluate the major parameters such as the dismantling schedule and cost, we created an equation to calculate them. Dismantling schedule consists of the man-power by unit of work, the number of workers, the basic unit time, and the number of such work times. Because there is no standard value in relation to a dismantling schedule during a reactor decommissioning, we take the range of the weighting factor into account to cope with an uncertainty of the parameters on a large scale.

An essential requirement in a decommissioning strategy of a nuclear facility is an evaluation of the radioactivity inventory. A calculation for a radioactivity inventory that can estimate an amount of radioactivity is very important as a means for improvement of the safety of a dismantling environment. In order to establish a hypothetical dismantling environment, the radioactivity distribution and location of the facility being dismantled should be designed in advance. We have conducted a visualization of the radioactivity distribution of facilities in KRR-2.

We completely visualized the radioactivity distribution of the objects by using a 3D contour mapping technology. The visualization of the radioactivity distribution that represents the level of radioactivity for an irradiated object laid the foundations to select the best scenario with the decommissioning analyzer and supervisor assigned to the dismantling procedure. Furthermore, it enabled the system to improve work efficiency and to call the workers attention to safety aspects. The results of the animation coincided well with the dismantling environment and the workers easily understood the scenario through the effect of multiple view points and collision detection. The results of a simulation that was performed based on the field data established that a nibbler process was the best procedure. Although the current system has significant errors as a result of using hypothetical data, our system is useful for planning and training by displaying the possible impacts of a decontamination and decommissioning for existing and future dismantling projects.

2.2.3. Treatment of irradiated graphite waste

In Korea, the irradiated graphite waste has arisen from the decommissioning of KRR-2. At the KRR-2, nuclear graphite was used as a moderator in the thermal column that was a space to irradiate an experimental specimen. The graphite waste has different characteristics from the other decommissioning radioactive waste due to its physical and chemical properties and also because of the presence of tritium and ^{14}C.

We are developing a number of technologies (Fig. 6) for the treatment of irradiated graphite waste and to demonstrate the developed technology. These include:

- Analysis of the physical, chemical and radiological characteristics of irradiated graphite,
- The technology for a release of Wigner energy and for the treatment of radioactive gas; and
- The technology for a handling, cutting and volume reduction of irradiated graphite.

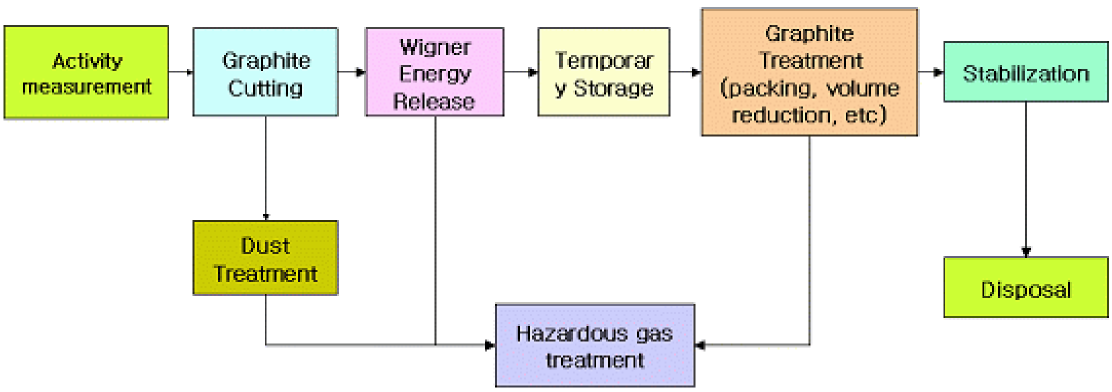


Fig. 6. Flow diagram of the irradiated graphite treatment for the decommissioning of KRR-1&2.

The radiological and physical properties of graphite waste from the decommissioning of KRR-2 have been investigated to develop the treatment technology for the irradiated graphite. It was found that various radionuclides are present - mainly ^{3}H, ^{14}C, ^{36}Cl, ^{60}Co, ^{134}Cs, ^{152}Eu and ^{154}Eu. The widely known effect of a density change in the irradiated graphite was observed in the experiment. The radioactive graphite has different characteristics from other radioactive wastes due to its physical and chemical properties and also because of the presence of ^{3}H and ^{14}C.

To quantify the stored energy in the irradiated graphite, the temperature of the graphite sample was raised in a controlled manner and the energy release rate was measured by DSC (differential scanning calorimeter) (Fig. 7). It was found that little stored energy was contained in the irradiated graphite from KRR-2. The stored energy of graphite samples was measured as 10 ~ 160 J/g and the energy release can be started when the heating temperature is raised to more than 120°C. The maximum release temperature is 200 ~ 250°C. The irradiated graphites from KRR-2 should be annealed at over 300°C to remove the stored energy for the purpose of their packaging, storage and ultimate disposal.

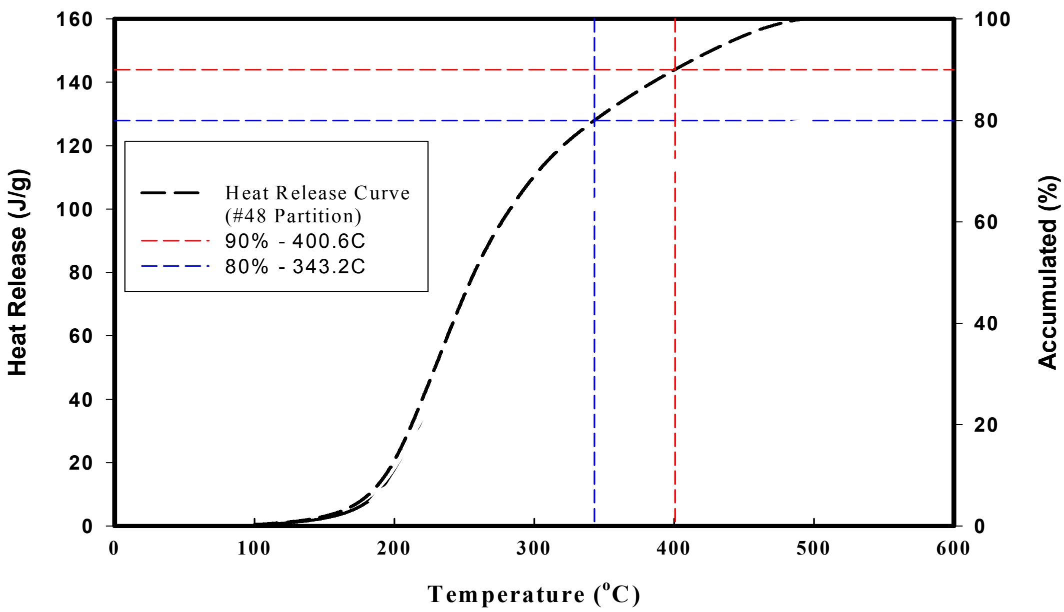


Fig. 7. Accumulated heat release during annealing of irradiated graphite from KRR-2.

Tested KRR-2 graphite waste was not easily destroyed in an oxygen-deficient condition. However, the oxidation reaction was found to be very effective in the presence of oxygen. No significant amount of the products of an incomplete combustion was formed even in a limited oxygen concentration of 4% O_2. It could therefore be positively said that an incineration would be very effective to reduce the volume of the KRR-2 graphite waste.

2.3. Decommissioning waste recycling technologies by a metallic waste melting

The decommissioning of nuclear installations results in large amounts of radioactive metallic wastes such as stainless steel, carbon steel, aluminum, copper, etc. It is known that the reference 1,000 MWe PWR and 881 MWe PHWR will generate metal wastes of 24,800 ton and 26,500 ton, respectively.

In Korea, the D&D of KRR-2 and an uranium conversion plant (UCP) at KAERI have been performed. The amount of metallic wastes from the KRR-1 and UCP is about 160 ton and 45 ton, respectively, up to now. These radioactive metallic wastes will induce problems of handling and storage from environmental and economic aspects. For this reason, prompt countermeasures should be taken to deal with the metal wastes generated by dismantling these retired nuclear facilities.

The most interesting materials among the radioactive metal wastes are stainless steel (SUS), carbon steel (CS) and aluminum wastes because they are the largest portions of the metallic wastes generated by dismantling these retired nuclear research facilities. As most of these steels are only slightly contaminated, if they are properly treated they are able to be recycled and reused in the nuclear field. In general, the technology of metal melting is regarded as one of the most effective methods to treat metallic wastes from nuclear facilities.

In aluminum melting tests, we found that ^{60}Co was removed and transferred to the slag and dust phases by the salt melting process. About 40 ~ 70% of ^{60}Co was removed from the aluminum ingot phase and partitioned to the slag phase and dust phase. ^{137}Cs was eliminated from the aluminum ingot phase and transferred to the slag and dust phases by up to 99%. About 20 ~ 50% of ^{137}Cs was partitioned to the slag phase and the remainder was mainly transferred to the dust phase. None of the cesium was found to be present in the ingots. Therefore we are planning to capture the cesium by a fly ash filter system. In the surrogate tests, the trend of the distribution for the surrogate nuclides was similar to that of the radioisotopes.

From the melting tests of the aluminum wastes taken from the KRR-2, the decontamination factors of the aluminum wastes are given in Table 3. Since cesium was not found by the MCA, there is no decontamination data for ^{137}Cs. As shown in Table 3, we found that ^{60}Co in the aluminum waste was easily decontaminated by using the melting technology.

Table 3. Decontamination factor (DF) for the melting of aluminum wastes from KRR 2

Sample type	Flux type*	Sample Activity	After Melting		DF
			Ingot	Slag	(Ingot)
		^{60}Co [Bq/g]	^{60}Co [Bq/g]	^{60}Co [Bq/g]	^{60}Co [Bq/g]
Al - Plate	Flux A	967.36	367.52	154.3	2.63
Al - Plate	Flux B	967.36	226.86	272.9	4.26
Al - Pipe	Flux A	190.61	48.13	3363.73	3.96
Al - Pipe	Flux B	85.96	17.8	1336.92	4.82

* Flux A: NaCl(45%), KCl(40%), Na_3AlF_6(15%), Flux B: NaCl(45%), KCl(40%), KF(15%)

In the melting tests of the steel waste (SUS, CS), we have investigated the partitioning phenomena of the ^{60}Co and ^{137}Cs radioisotopes in the ingot, slag and dust phases by using various slag types, slag concentration and basicity in an arc melting process. Most of the ^{60}Co remained in the ingot phase, while it was barely present in the slag phase during the steel melting. ^{60}Co decontamination factor was not highly dependent on the slag composition. However, it was found that a highly fluid basic slag former is somewhat effective. The distribution ratio of ^{60}Co into the ingot and the slag phase showed that about 90% to 95% was recovered in the ingots. ^{137}Cs was completely eliminated from the melt of the stainless steel as well as the carbon steel and distributed to the dust phase. The portion remaining in the slag phase depended considerably on the slag basicity. A maximum of 65% of the ^{137}Cs remained in the slag phase with a high slag concentration and basicity. Generally, ^{137}Cs distribution in the slag phase was between 10% and 25% during lab-scale arc furnace testing. In the real steel wastes generated from KRR-2, the trend of the radioactive nuclides such as the cobalt and cesium was similar to that of the radioisotope (^{60}Co and ^{137}Cs) tests.

We have a plan to perform a pilot scale decontamination test for the melting of β and α contaminated metallic wastes. The treatment capacity of the pilot plant is 200 kg/hr for non-combustible wastes. The phases of the pilot scale melting test consist of a functional test and system stabilization, a melting of the metallic wastes, and followed by analysis and evaluation.

3. Conclusions

KAERI has carried out innovative technology development in the fields of dry decontamination, decommissioning and decommissioning waste treatment to support its on-going decommissioning project as well as hot laboratory maintenance, and obtained the following conclusions:

3.1. Dry decontamination technologies for hot laboratories

As a PFC solution is a non-conducting material and easily separated from contaminants, a PFC spray decontamination process is a promising method to decontaminate surfaces loosely contaminated with radioactive particles.

A good decontamination performance was demonstrated by means of a lab scale hot-test with radioactive specimens and the IMEF hot cell test.

Both the PFC and CO2 pellet blasting decontamination technologies are considered to be applicable to hot cells including DFDF (DUPIC Fuel Demonstration Facility) for a maintenance and refurbishment of them in the future.

3.2. Decommissioning technologies for the research reactor decommissioning

The Monitoring System for a Simultaneous Measurement of Alpha- and Beta Contamination in a Pipe and a digital mock up system are applicable to the decommissioning of the KRR-1&2 and Uranium Conversion Facility in Korea and also to the in-situ measurement of embedded pipes at operating nuclear facilities in the future.

The irradiated graphites from KRR-2 are to be annealed at over 250°C during 60 min to remove their stored energy for the purpose of packaging, storage and ultimate disposal. Incineration by fluidized bed is thought to be a good choice to reduce the waste volume.

3.3. The decommissioning metallic waste melting technology

In the aluminum melting tests, cobalt was captured at up to 75% into the slag phase. Most of the cesium was completely eliminated from the aluminum ingot phase melt and moved into the slag and dust phases.

In the melting of steel, ^{60}Co was almost retained uniformly in the ingot phase, whilst most of the ^{137}Cs was partitioned between the slag and the dust phases in the off gas.

ACKNOWLEDGEMENTS

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PROJECT MANAGEMENT SYSTEM FOR DECOMMISSIONING OF RESEARCH FACILITIES

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Abstract
At the Korea Atomic Energy Institute (KAERI), two research reactors (KRR-1 and KRR-2) and one uranium conversion plant (UCP) are being decommissioned. In 1997, a project was launched for the decommissioning of these reactors with the goal of completion by 2008. A separate project for the decommissioning of the UCP was launched in 2001. The physical dismantling of works was started in August 2004 and the entire project will be completed by 2009. KAERI has developed a computer information system, named DECOMMIS, for project management with an increased effectiveness for decommissioning projects and for record keeping for future decommissioning projects. The management system consists of three sub-systems; code management system, data input system (DDIS) and data processing and output system (DDPS). Through the DDIS, the data can be directly input at sites to minimize the time gap between the dismantling activities and the evaluation of the data by the project staff. The DDPS provides useful information to the staff for a more effective project management and this information includes several fields, such as a project progress management, manpower management, waste management, radiation dose of workers and so on. The DECOMMIS system was applied to the decommissioning projects of the KRR and the UCP, and utilized to give information to the staff for decisions regarding the progress of the projects, to prepare the reference data for the R&D for the development of planning tools, and to maintain the decommissioning data for the next projects. In this paper, the overall system will be explained and several examples of its utilization, focused on a waste and manpower control, will be introduced.

1. Introduction

1.1. Facilities decommissioned

At KAERI, two research reactors (KRR-1 and KRR-2, Table 1) and one uranium conversion plant (UCP) are being decommissioned. The first research reactor in Korea, KRR-1, was a TRIGA MARK-II type (open pool and fixed core), and its power was 100 kWt at its construction and it was up-graded to 250 kWt by KAERI. The detailed characteristics of the research reactors are shown in Table 1. Its construction started in 1957, and its first criticality was reached in 1962 and it had been operated for 36,000 hours until its final shut down in 1995. It played a role in the development of the basic nuclear science and technologies and for an increase of the nuclear industries in Korea. The second reactor, KRR-2, was a TRIGA MARK-III type with an open pool and a movable core and its power was 2 MWt. Its first criticality was reached in 1972 and it had been operated for 55,000 hours until the decision to decommission it in 1995. The main purpose of the KRR-2 was the production of radioisotopes and neutron utilization research, such as neutron radiography.

The UCP is located at KAERI, in Daejeon, and it was constructed for development of the manufacturing technologies and the localization of nuclear fuels production in Korea. Main product of the facility was uranium dioxide powder, which was bound for the fabrication plant for the fuels of a heavy water reactor, Wolsung #1. The commissioning was finished in 1993 with the ADU (Ammonium Di-Uranate) process but the first production was completed in 1998 with the AUC (Ammonium Uranyl Carbonate) process. The capacity of the facility was 100 tons of uranium per year but it was too small to be economically feasible. Finally it was determined to shut it down in 1993, after the production of 320 tons of UO_2 powder.

Table 1. Characteristics of the KRR-1 & 2

Items	KRR-1	KRR-2
Reactor Type	Open pool, Fixed core	Open pool, Movable core
Thermal Power (kW)	250	2000
First Criticality	1962. 3. 19	1972. 5. 10
Shut down	1995. 1	1995. 12
Total Operating Time (Hours)	36,000	55,000
Total Generating Power (MWh)	3,700	69,000
Neutron Flux ($n/cm^2 \cdot sec$)	1×10^{13}	7×10^{13}
Fuel		
Contents of U (w/o)	8.5	8.5
Enrichment (w/o)	20	70
Cladding	Al	304SS
Chemical composition	$U\text{-}ZrH_{1.0}$	$Er\text{-}U\text{-}ZrH_{0.6}$
Moderator/ Coolant	H_2O	H_2O
Reflector	Graphite	H_2O
Control rod	B_4C	B_4C

1.2. Decommissioning Project

In 1996, it was concluded that KRR-1 and KRR-2 would be shut down and dismantled. A project was launched for the decommissioning of these reactors in January 1997 with the goal of completion by 2008. The total budget for the project is US$20M, including the cost for the waste disposal and for the development of the technologies. The work scopes during the reactor decommissioning project are the dismantling of all the facilities and the removal of all the radioactive materials from the reactor site. After confirming the removal of the entire radioactivity, the site and buildings will be released for unrestricted use. A separate project for the decommissioning of the UCP was launched in 2001. This project will be completed by 2009 and the total budget is US$10M. The detailed schedules of the projects are shown in Figure 1.

For the decommissioning projects, KAERI constructed a center (Decommissioning Technology Development Center), which was divided into two functional groups: project group and R&D group. The project group was again divided into two project management departments and each department has several sections according to the work areas as shown in Figure 2. Manpower for the different sections was supplied from different companies and the data for the decommissioning activities were independently input by each section. The central organization of KAERI was utilized as far as the groups of radiation protection and quality assurance are concerned, which satisfies the legal requirements for an independency of the quality and safety activities.

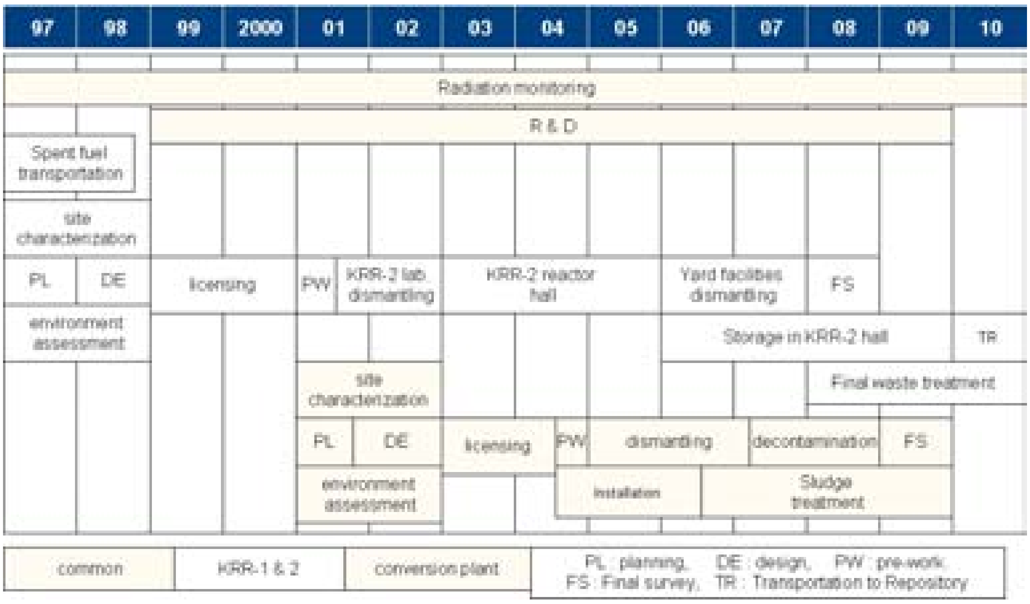

Fig. 1. History and plan of the decommissioning projects at KAERI.

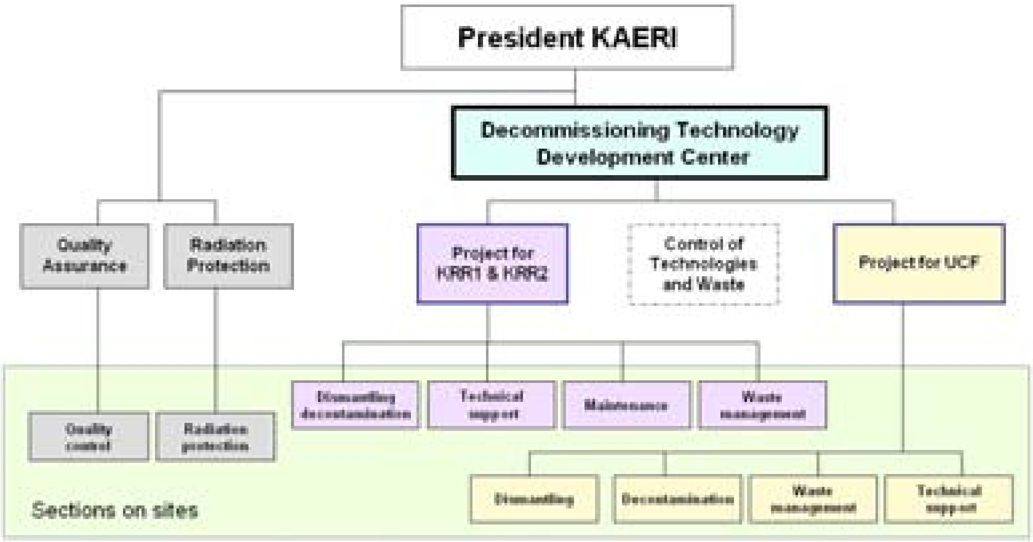

Fig. 2. Organization for the decommissioning projects at KAERI.

1.3. Decommissioning Activities

The first facility to be decommissioned was the auxiliary facility of the KRR-2 which consisted of 12 laboratories, 10 lead hot cells and 2 concrete hot cells which were used for experiments with radioisotopes. In the 12 hot laboratories all the apparatus and installed equipment was dismantled to reduced size with hand tools such as saws and cutters. After the removal of all the dismantled pieces, the ceiling, wall and floor were decontaminated with abrasive papers, movable vacuum cleaners, scabblers, and grinders. For the 10 lead hot cells, the pipes and electric wires were disconnected, and then the rear doors, lead bricks, glass windows, and finally the concrete structures were dismantled and removed. In the concrete cells, the dose rate was too high, due to the irradiated samples, to dismantle the objects inside the hot cells. Because of a lack of records and opaque window glasses from browning, the first work for the dismantling was an identification of the dismantling objects in the cells with a gamma camera and a remotely operated video camera [1].

The core structure of the KRR-2 was also dismantled, cut into small pieces and packed into a shielded waste cask under water. The rotary specimens rack (RSR), inserted into the reactor core like a ring, was separated and moved to the pool of the KRR-1 to dismantle it with a specially developed tool. Besides the reactor core in the pool, there were many pipes and ducts, for radiation of samples and 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 [2].

From May 2005, the dismantling of the bio-shielding concrete of KRR-2 started, and completed at the end of November 2005 [3,4]. Before the main cutting works for the shielding, 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 graphite blocks were removed from the thermal columns. The blocks, located near the core, were much more activated than expected and a remotely operated gripping tool was developed and used for pulling the graphite blocks out. The stainless steel pipes of the beam ports and the concrete near the pipes were highly activated by neutrons. A boring machine was used to remove the beam port pipes and the concrete around the pipes at the same time. The first work for the characterization of the concrete was to gain an understanding of the exact shape and size of the shielding and the second was for a measurement of the physical properties such as the density. Finally, a matrix sampling on the surface and samplings along to the depth of the concrete was carried out. From the measured radioactivity of the samples, mapping of the surface radioactivity was carried out and it was extended to 3 dimensional diagrams with a general dependency of the radioactivity along the depth. With the results of this pre-work and characterization, the sequence of cutting and the cutting machines were determined and detailed cutting procedures were established. Also the cutting lines were determined by considering the capability for taking-down machines to the work-face (the existing crane and a fork lift), accessibility of a diamond wire cutting saw and transportation vehicles, and the margins for separation between an activated and a not-radioactive part. 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 m^3 waste containers.

At the decommissioning of the UCP, the decommissioning plan was approved in July 2004 by the MOST (Korea Ministry of Science and Technology), and then several preparative works, such as the preparation of detailed work procedures, the installation of new utility systems, and the establishment of an analysis system for low alpha radioactivity, were carried out. Also a preliminary cleaning was carried out to reduce the air contamination due to the re-suspended dust from that deposited during 10 shut-down years. The dismantling and removal of the equipment started from a kiln room, the most remote room from the waste discharge gate. The ceiling, wall and floor were decontaminated with a steam jet device and a scabbler, and the dismantled metal parts were decontaminated to reduce the radioactivity to below 0.2 Bq/cm^2 with steam jet cleaners, ultra sonic/chemical decontaminators and an electric polisher [4].

1.4. Decommissioning Strategies

The following strategies were chosen at the beginning of the preparation step for the decommissioning project of the research reactors and they were extended to that of the uranium conversion plant.

- Dismantling time: immediately after the decision

- Final state of the site: free release of the site and buildings after a removal of all the radioactive materials for an unrestricted use.

- Waste: minimization of the solid wastes, which will be packed and sent to the national repository facility, and a near zero release for liquid waste.

- Technologies: development of the technologies required for the dismantling of the facilities in the projects and for any future demands.

- Participation of commercial companies and a technology transfer to them.

Above all, safety is the first priority in the strategy for the projects. Safety includes low radiation dose to the workers and protection of the environment and the public around the sites. This safety aspect was evaluated in the design stage of the decommissioning plans and reviewed and approved by KINS (a consultant organization of the regulatory body) [5].

2. Project management system (DECOMMIS)

Basically, the decommissioning data was managed with three sub-systems: a code management system, a decommissioning data input system (DDIS) and a decommissioning data processing/output system (DDPS) as shown in Figure 3. By code management system is meant symbols or abreviations for inputting data to the DECOMMIS program. All the data on the dismantling activities was input at the sites through the DDIS and the data was processed in a simplified and formatted manner to provide useful information to the staff of the decommissioning projects and the reference data to the R&D members [6]. All the input data was duplicated in a back-up system.

2.1. Objectives of the DECOMMIS

During the progress of the projects at KAERI, a revision of the selection of the dismantling methods and change/termination of the dismantling activities were undertaken many times. For an early and adequate decision, information on the ground of the site activities was essential, and for correct information, a project management system, which was a kind of computer information management system, was developed.

All the solid waste generated from the decommissioning sites was classified into three groups [7]. The first group was the not-contaminated waste, and its radioactivity was less than 0.013 Bq/g for its beta/gamma activity. The second group had a higher activity than 0.4 Bq/g and was packed into drums of 200 liters or into containers of 4 m^3, to be sent to the national repository plant. The third was the conditional releasable waste, and the waste with a radioactivity between the non-contaminated (the first group) and the radioactive waste (the third group) can be classified into this group. KAERI will be allowed to treat it along a predetermined route after a sufficient level of safety has been approved, by the regulatory body.

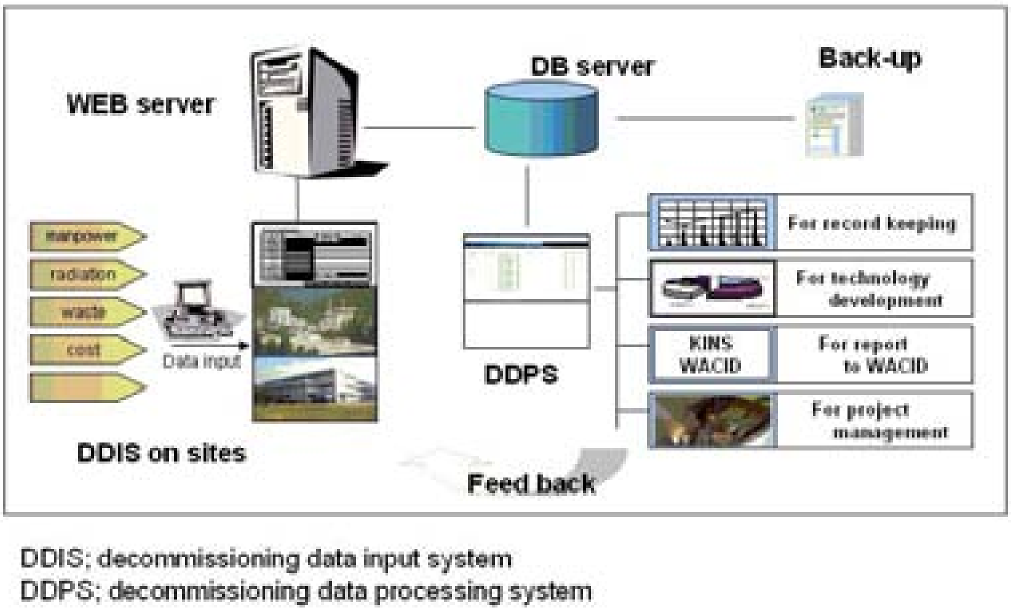

Fig. 3. Conceptual diagram of the structure of DECOMMIS.

The minimization of the radioactive waste, which is one of the identified strategies, could be achieved by the repeated decontamination up to the lowest practical level. An example of waste minimization activity was the iterative decontamination of the concrete structure pieces generated during dismantling of the 10 lead hot cells, which was carried out from October 2001 to May 2002. A total of 670 man-days (MD) were required for the dismantling and the decontamination, but the estimation during the planning was 474 MD. It was evaluated that these increased requirements of the man-power were due to the repeated measurements of the very low radioactivity of 0.013 Bq/g and the waiting time between the decontamination works. By consuming this manpower, only 184 kg, among 32 tons of the concrete waste, were classified as radioactive waste. This means that the amount of radioactive concrete waste was considerably reduced and also the disposal cost was considerably decreased. But the project period was extended and therefore the cost for the manpower and the administration was increased. As a result, there was an increase of 0.8 % of the total budget during the dismantling of the lead hot cells because of the increased time period. It was believed that there is an optimum point to terminate the decontamination work of the concrete blocks but this decision point was missed because there were no tools for data evaluation.

2.2. Soft and hard ware

The minimum software and hardware requirements for the system environment are shown in Table 2 [8]. As shown in the table, the hardware PC grade computers with Pentium CPUs. The software, “ORACLE,” operated on WINDOWS, was selected because it is widely used at KAERI for internal communications. The system was designed to operate on the internal LAN of KAERI and to input the data at the PC of the section heads, but because the decommissioning site of the KRR-1 and KRR-2 is located in Seoul, data files were made and transferred to the system manager by e-mail to add the data to the main DB server.

Table 2. Minimum requirement for system environments

ITEM	SYSTEM		MINIMUM REQUIREMENTS
Software	Server	Browser	Internet Explorer 5.5
		DB server	ORACLE (RDBMS 9i)
		WEB server	ORACLE 9i AS
		OS	Window 2003 Server
		Middle ware	SQL Net
	Client	Browser	Internet Explorer 5.5
		OS	Window 98
Hardware	Server	DB/WEB server	CPU: Pentium 4, 2 CPU HDD: 100 GB / Memory : 2 GB
	Client	Client PC	CPU: Pentium series HDD: 80 GB / Memory : 1 GB

2.3. Considerations for design of DECOMMIS

Before the design of the DECOMMIS, the basic requirements were established. The main requirements of each system and their measurements to meet them are summarized in Table 3. One of the most important requirements of the input system is to input the correct data. Incorrect data could be caused from errors in typing and interpretation differences between persons who input the data. A maximized utilization of codes in a formatted (tabulated) input space and a minimization of the text input were selected as measures to meet these requirements. Another important requirement is a short time lag between the dismantling activities and the evaluation of the data so as not to miss the chance of proper management. For this, direct data input by the individual workers at the site immediately after their work was considered, but the capability of the workers to use the computer was not so high and they were not reliable in their documentation. Therefore ‘input by a section head’ was selected as a measurement variable. To be able to input data from an existing file is also selected as a requirement for the DECOMMIS. The development of the system started after the project had been started and some data was already summed up, but in different formats. Firstly, all the existing data were converted to Excel files, which can be directly input to the system. For the output system, tables of the data, sorted according to the search conditions, were selected as an output format. An output in figures and graphs was also considered because they seemed to be easily understood, but tables were selected for flexible utilization of the data.

All the data from a decommissioning site was categorized and input into several data fields, which were chosen by considering the work characteristics and the kinds of data to be input. The data from each section of the organization (Fig. 2) was to be input into different data fields by the section head. Sometimes, many sections participated in one decommissioning activity and the section heads had to individually input their own data into their fields. Therefore, a connection between data fields was necessary to combine the data distributed in many fields. For example, the waste management section, decontamination section, radiation protection section and QA section involved in the packaging of the radioactive waste into a 4 m^3 container, which was continued for a month. In order to evaluate the manpower consumption for packing the wastes into the containers, all the man hours of each working day and of each section should be summed. In this case, a connection between the waste data field and the manpower data field was necessary to sort the manpower data for a waste container.

For this, a work serial number was defined for every specific work and all the data was input under the serial number even for different data fields as shown in Figure 4.

Table 3. Requirements for the DECOMMIS

SYSTEM	REQUIREMENTS	MEASUREMENTS
Data input	- Purpose of DECOMMIS - Correct data - Fast input immediately after works - Mentality and capability of workers - Utilization of existing data as files - Efficiency	- Various data - Input as codes - Input by section head - PC-based and easy software - Data input from Excel files
Data processing	- Connection between data fields	- Work serial number
Data output	- Easy search - Flexible and widely applicable data - Avoid duplication of works - Not easy access to KAERI network	- "Wild card" letter - Intermediate data as table - Easy data transfer to Excel file - Printing output to reports

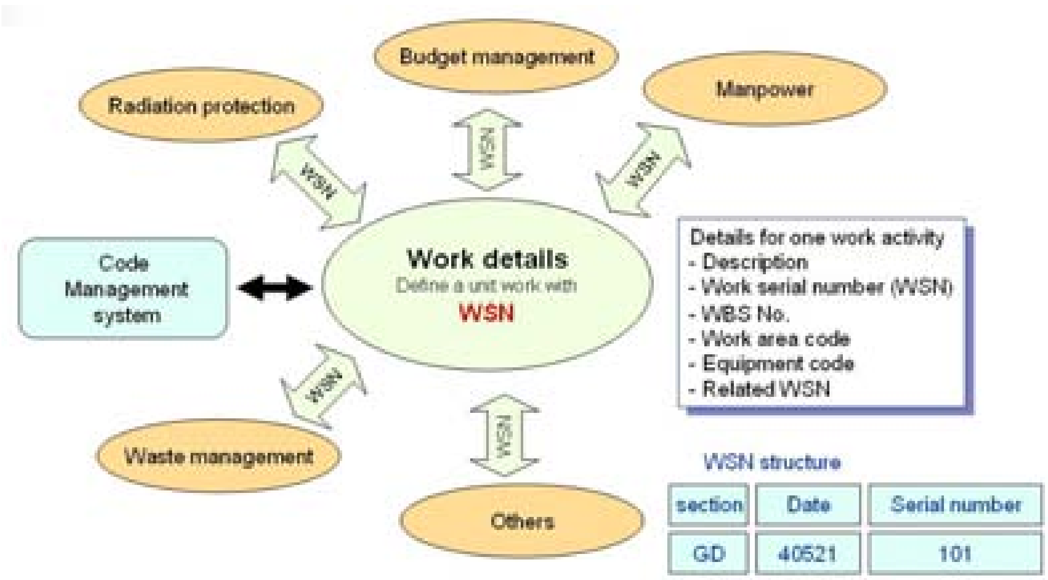

Fig. 4. Role of the WSN as a connection tool between data fields.

2.4. Decommissioning data input system (DDIS)

The kinds and the classifications of the data to be input into DECOMMIS are shown in Table 4. The data were divided according to the periodicity of the data generation; e.g. daily and non-periodic generation. The waste packaging and placement into the drums and containers was carried out two or three times a month and the data about the waste drums or containers were generated and input at the date of generation. Therefore, the data on the waste drums was classified as non-periodic data. For record keeping purpose, the photos of the dismantling activities, facilities and wastes, and all the technical documents, such as the decommissioning plan, procedures and manuals and presentation materials at a conference, were also input into DECOMMIS.

Table 4. Kinds of data input to DECOMMIS

WORK AREA	DAILY GENERATION	NOT PERODICAL GENERATION
Work details (D&D)	Name of project Summary and remarks Detail description of all activities	Photos of works
Project	Manpower consumption	Cost output
Radiation protection	Personal exposure dose Monitoring of working condition Surveillances of space activity	Surface contamination Work permission
Waste management	Solid waste Liquid waste Ventilation system operation	Definition of drum/container Characteristics of waste Decontamination work record
QC	QC activities	
Technical support	Technical supporting activities	Technical documents
Common	Equipment management Internal/external training	

2.5. Decommissioning data processing system (DDPS)

The data, corrected through the DDIS system, will be processed by the DDPS to show adequate variables for the management process. The names of the output variables and their purposes are listed in Table 5. It will be very important to select the kind of data in each variable because the selected data could provide the basic information for an exact understanding of the progress of the projects and for a determination of the points to change the project steps. A data processing consists of four steps: the input of requirements, the data sorting, the data processing, such as the summation of group data, and the data formation into a desirable format including its transformation into other files.

The methods and variables to be input and processed for a project management system could be different from those of other nuclear facilities. The data on uranium contamination in air was more important at the UCP decommissioning site, while it was the spatial dose rate at the KRR site. It is understood that these differences came from the site dependent conditions, such as the kinds of facilities, the scale of the projects, the selected strategies and the conditions of the industries.

Table 5. Kinds of data to be processed by DECOMMIS

AREAS	NAME of VARIABLES	PURPOSE
Project	Contract	Manpower calculation for contract payment
	MP-analysis-cond	Manpower consumption for a specific condition
	MP-analysis-area	Manpower consumption for a specific area
	MP-analysis-work	Manpower consumption for a specific work
	Individual	Time distribution of an worker
	Progress	Project progress
Waste management	WACID	Data generation for WACID DB
	Amount	Waste generation (weight/activity)
	Trace	Tracing of a waste piece to package
	Decon	History and efficiency of decontamination
Radiation protection	Exposure	Exposure dose of a worker or a group
	Surveil (Ventil)	Space dose rate (radioactivity of stack air)
	Surface	Surface contamination of working area
	Expectation	Comparison of expected and exposure dose
Budget	Overall	Overall cost
	Account	Cost according to account numbers
	Item	Cost for a kind of items
Others	Equipment	Utilization and maintenance
	Training	Information on training/education
	Photo	Photographs
	Tecdoc	All document

3. Utilization of DECOMMIS

3.1. External report

The data for a dismantling waste generation was prepared from the DECOMMIS for the report to a national radioactive waste database, WACID (WAste Comprehensive Information Database) [9], operated by KINS (Korea Institute of Nuclear Safety). When a period required for the report was input, all the necessary data on the radioactive wastes generated during that period was sorted and modified to fit the required forms for the report. The final results were converted into an Excel file which could be directly transferred to WACID and printed as hard copy in order to confirm it off line.

3.2. Change of Contract

As indicated, the shielding concrete was dismantled from May to November 2005. For the project of the KRR decommissioning, a main contract for the supply of the manpower was made with a company, named Doosan Heavy Industry which made many sub-contracts with different companies for technology support, radiation protection, dismantling/decontamination and waste management. This contract system was selected for easier control of the manpower and fewer contract numbers for KAERI. Due to this contract system, the workers were fixed; which carried out all kinds of work during dismantling. For the first 6 weeks (first period in Figure 5) of the concrete cutting, the workers from the contracted companies carried out the concrete cutting with a small capacity wire saw supplied by KAERI. Because of their low abilities, the cutting rate was less than 10 tons/week and it was thus expected to require more than two years to complete the cutting. A new contract was made to supply man-power from an experienced company and the cutting by the experts was continued for 5 weeks with the wire saw supplied by KAERI. The cutting speed was increased to about 40 tons/week but it was still not satisfactory. Finally, a third contract for the lease of large capacity wire saw machines was made and the concrete cutting was carried out by experts and with these large machines. The cutting speed was increased to 120 tons/weeks and the dismantling of the not-radioactive concrete could be completed within 5 and half months [10]. Requirements of these new contracts were determined by the waste and manpower consumption data provided by DECOMMIS.

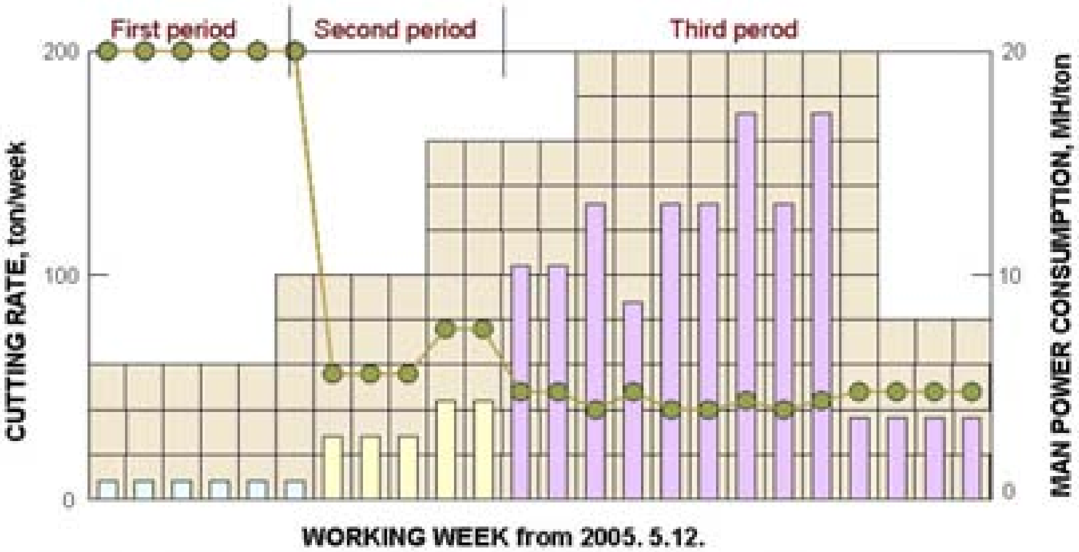

Fig. 5. Change of cutting speed of KRR-2 concrete shielding at three different contracts.

3.3. Control of manpower consumption

In early 2005, it was found that the dismantling works at the decommissioning site of the UCP were delayed by 3 months beyond the schedule prepared at the decommissioning planning stage. The delay was caused by an under-estimation of the required manpower for dismantling of the nuclear chemical plant. The Korean standard for dismantling chemical industries in general was used as a basis for an estimation of the manpower. Two workers were added to the waste management section in April and two or three workers of the waste management section were asked to assist in the dismantling work. Also, the decontamination works were temporary stopped and the workers of the decontamination section also assisted in the dismantling works. After 5 months, the schedule was nearly caught up, but un-treated metal pieces started to accumulate in the dismantled rooms of the conversion plant. On September, manpower was finally adjusted again as shown in Figure 6: two workers were permanently transferred from the decontamination section because they had become skillful in the dismantling section and two new workers were added to the decontamination section. After that, manpower consumption of each section became stabilized and little assistance from other sections was required. This arrangement of the manpower was done with the help of DECOMMIS data.

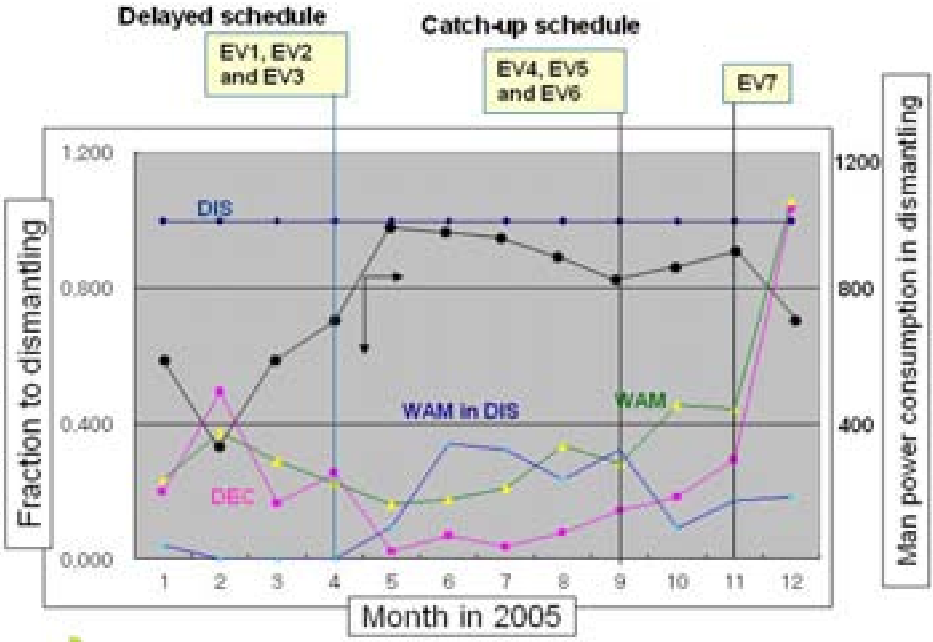

Fig. 6. Manpower consumption of a dismantling area during the UCP decommissioning.

Where EV1; Temporary stop of decontamination work
EV2; Increase 2 workers in WAM
EV3; Dispatch 2 or 3 workers of WAM for assistance DIS
EV4; Restart of decontamination
EV5; Permanent transfer of 2 workers from WAM to DIS
EV6; Return of 1 worker to WAM
EV7; Increase of 2 workers in WAM and start of operation of sludge treatment
And WAM ; waste management section
DEC; decontamination section
DIS; dismantling section

3.4. Assessment of economic benefit on a repeated decontamination of metal pieces

One of the strategies in decommissioning projects is a minimization of the solid radioactive waste. To meet this strategy, a repeated decontamination was carried out until the radioactivity of the dismantled pieces was decreased to under the limit of a radioactive waste, 0.4 Bq/g. By this repeated decontamination, a minimization of the waste could be achieved but it was expected that more cost could be incurred for the decontamination. An evaluation was carried out for two dismantling objects [11]: an aluminum duct (case A) in the pool of KRR-2 and a stainless steel storage rack (case B) for the beam port plug. They were cut into small pieces and decontaminated by an ultrasonic cleaner which was developed by KAERI. When the residual radioactivity of a decontaminated piece was higher than the limit, the piece was decontaminated again. According to the repeated decontamination, the weight of the radioactive waste was decreased but the total cost (treatment cost + disposal cost + administration cost) was increased or decreased. For an easily decontaminated component such as the rack, a decrease of 60% of the required cost was achieved while there was a 20% increase for the aluminum duct (see Figure 7). This increase of the cost was accepted for KAERI's projects because the strategy foccused on waste minimization.

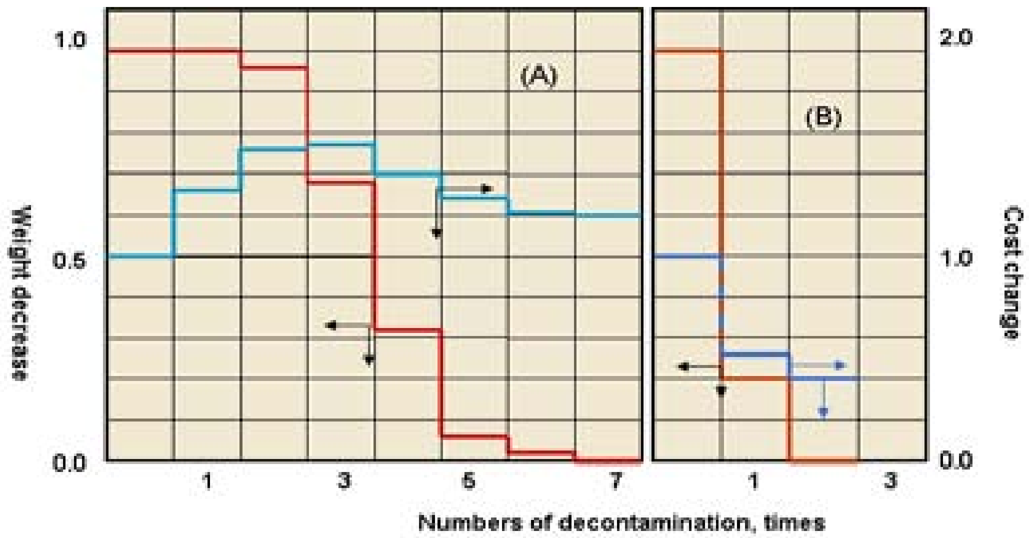

Fig. 7. Change of the weight of radioactive waste and cost for treatment.

3.5. Estimation of manpower for a decommissioning project

From the results of the evaluation of manpower consumption, a method for the estimation of the total manpower (M_T) is proposed using the following equations [12]:

$$(1+E_r)M_r = \sum_I W_i R_i$$

$$M_T = \sum_R M_{r,dis}\left(1+\sum_K f_k\right)$$

Where W_i ; is the total manpower required to dismantle an object i

Ri ; Standard manpower required per unit of measure of the object type I (e.g. hours/Kg)

$M_{r,dis}$; Required manpower for dismantling a facility of group r

E_r ; Easiness factor (corresponds to a "difficulty factor" of $\frac{1}{1+E_r}$)

Here f_k means the ratio of the manpower requirement in work section k to that in the dismantling section. That is, putting the manpower of the dismantling section as 1, f_{RP} is the required manpower of the radiation protection section and various f values of the KRR and the UCP dismantling are shown in Table 6. In the case of the KRR, f values of all the sections have two patterns: for dismantling the concrete structure and for the other facilities. For the UCP decommissioning, the pattern of the f values can be classified into three: higher, normal and lower manpower requirements. A room with large and heavy equipment where the powered uranium compound was handled, such as a kiln room, was categorized into a room that required a higher manpower for a dismantling.

Table 6. Various f values for the KRR and the UCP decommissioning sites

<table>
<tr><td>Room or facilities</td><td>ME/CU(3)</td><td>DT</td><td>RP</td><td>QA</td><td>TC</td><td>CW</td></tr>
<tr><td>KRR</td><td>0.54</td><td>0.55</td><td>1.15</td><td>0.14</td><td>0.24</td><td>0.45</td></tr>
<tr><td>KRR (bio-shielding concrete)</td><td>0.24</td><td>0.36</td><td>0.46</td><td>0.10</td><td>0.15</td><td>0.23</td></tr>
<tr><td>UCP (higher manpower)</td><td>0.73</td><td>0.83</td><td rowspan="3">(1)</td><td rowspan="3">0.11</td><td rowspan="3">0.15</td><td rowspan="3">(2)</td></tr>
<tr><td>UCP (normal)</td><td>0.33</td><td>0.55</td></tr>
<tr><td>UCP (lower manpower)</td><td>0.15</td><td>0.28</td></tr>
</table>

Note (1) under evaluation

(2) Manpower is included in each area.

(3) ME; maintenance at KRR and CU cutting for decontamination at UCP site

4. Interaction with other CRP members

An Argentine expert (Mr. Silvio Alejandro FABBRI), from Comisión Nacional de Energía Atómica, visited the decommissioning sites at KAERI in Seoul and Daejeon from 12th to 14th September, 2005. The status of KAERI decommissioning projects of the KRR-1 and -2 and the UCP, and a R&D program on decontamination in NAEC of Argentina was introduced and an on-job participation of an Argentine expert to the KAERI decommissioning projects was agreed. According to the agreement, another Argentine expert (Mr. Gabriel Raul RUGGIRELLO) participated to the decommissioning project of the uranium conversion plant from June 5th to July 31st 2006 (for 2 months). His main job was the characterization of the experimental facilities for UF_4 production which was a pilot plant and will be dismantled from 2009.

After the first CRP meeting in Halden, there were many discussions between Brazilian participant (Mr. Paulo Ernesto DE OLIVEIRA LAINETTI, from Instituto de Pesquisas Energéticas e Nucleares IPEN) and a KAERI member at the meetings and through e-mails. The topics of discussions were a confirmation of an umbrella agreement on nuclear energy between Brazil and Korea, difficulties of dismantling techniques and waste management on the decommissioning of nuclear fuel manufacturing facilities and possible cooperation between IPEN (Brazil) and KAERI (Korea). Both sides agreed to do their best to invite experts to their decommissioning sites in 2008.

The presentation of Austrian participant (F. MEYER from Nuclear Engineering Seibersdorf GmbH – NES) contained the dismantling experience of shielding concrete cutting by a diamond wire saw and clearance of waste concrete. Some data on management techniques of cooling water from cutting of concrete shielding and the Austrian clearance level for waste release were requested by the KAERI member. The idea of the management techniques of cooling water was utilized and modified for gathering and recycling the cooling water from cutting of concrete shielding of KRR-2. The data on the clearance level were utilized as one of the references for the assessment of the clearance of the waste concrete from KRR-2.

5. Conclusion

During the progress of the decommissioning projects, a decision on, or a change of, the decommissioning activities was inevitable even though a better decommissioning plan and an earlier/better decision might have increased the efficiency of the projects. During the projects at KAERI, selection of the decontamination process, manpower control and change of equipment were all experienced. For improved decisions, a project management system, named DECOMMIS, was developed. Its function was extended to record keeping for the preparation of future projects, data analysis for R&D and data sorting for reports on wastes from sites to a national waste database.

The system is a kind of information management system and it consists of a data input system (DDIS) and a data processing/output system (DDPS). At the design step of the DECOMMIS, the correct input, the existing data files, the capability of the workers to use a computer, a work serial number for the connection between data fields, and an easy search and flexible application for the DDPS were considered as the basic requirements. Now, the decommissioning data from the KRR-1, the KRR-2 and the UCP sites is being gathered.

Using the DECOMMIS system, periodic reports to the WACID DB were prepared and a new contract for a concrete cutting and manpower management for the UCP decommissioning was possible. Also, by an evaluation of the waste amount and the required cost, it was found that an increase of the cost by repeated decontamination was acceptable for the projects. These mean that DECOMMIS can play a strong role as a project management tool. From the results of the assessment of the manpower consumption characteristics, a manpower estimation tool was proposed for the next decommissioning projects. DECOMMIS will provide the proper data on the ground for physical dismantling works and these proper data enable a manager to select an optimum contract, a better dismantling device and termination points for waste decontamination.

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DEVELOPMENT AND SELECTION OF DECONTAMINATION TECHNIQUES FOR DECOMMISSIONING PROJECTS IN THE RUSSIAN FEDERATION

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Abstract
There are a number of contaminated sites as a result of the development of nuclear weapons and nuclear energy in the former Soviet Union. [1,2]. There are a lot of activities needed to be carried out for remediation of these sites. One of the biggest tasks is decontamination of process equipment, constructions, and land. Next is development of decontamination plans which will respond to all safety requirements and will consist of detailed information, techniques, tools and procedures needed to carry out decontamination and/or demolishing works and to reach safe limits for actual sites. This decision-making process may be based on knowledge about available decontamination techniques, its advantages, disadvantages and limitations. All recommendations should be based on real practice and include knowledge of technical safety and organizational tasks.

1. INTRODUCTION

Decontamination is removal of radioactive contaminants from surfaces of metal, alloys, polymers, paints etc. The decontamination efficiency is characterized by the decontamination factor (DF) expressed in terms of the ratio of the initial level of radioactivity to the residual level achieved through a decontamination process.

The choice of decontamination process depends on specific equipment design, the character of contaminated material, the level and conditions of contamination and the decontamination level desired [3-7]. An optimum decontamination technique selected provides the maximum cost-effective removal of contaminants with a minimum of radwaste and personnel exposure. Since the middle 60s, decontamination purposes traditionally have used aggressive solutions. A great number of physical, mechanical and chemical decontamination techniques and equipment are available all over the world and in Russia at present time.

Since the mid-1960s, research institutes of the former USSR have been developing decontamination techniques for a variety of materials and contaminants for nuclear fuel cycle and former nuclear weapon facility needs [8-12].

Firstly, in the development, chemical decontamination was the most commonly used method. According to the nature of contaminants and contaminated material, mineral acids, alkali, mineral and organic oxidants and reductants were used. For best results, complex-forming agents were sometimes added. However, in spite of widespread use of chemical decontamination at the USSR nuclear facilities, this technique has a drawback of producing a great deal of secondary liquid radwaste. Since the early 1970s, attention has focused on the reduction of radwaste. Currently, optimized electrochemical and strippable coating methods are showing the greatest promise.

2. SCIENTIFIC BACKGROUND OF DECONTAMINATION

Decontamination is the removal of radioactive contaminants from the contaminated surface. The Decontamination factor (DF) is usually used for the decontamination efficiency characterization. It is expressed as the ratio of the radioactivity level before (Ai) and after (Af) decontamination.

$$DF = Ai/Af \qquad (1)$$

In Russia these factors are also characterized by:
- residual (final) activity, A_k;
- decontamination factor (A_n/A_k), where A_n is the initial contamination level;
- decontamination index (D_d); $D_d = \lg A_n/A_k$;
- decontamination extent (% of activity removal) or the proportion of contaminants removed (%)

$$\beta_d = A_n - A_k/A_n \,.\, 100\%; \qquad (2)$$

- percentage of residual contaminants remaining on the surface (%) $\alpha = A_k/A_n\ 100\%$;
- dose-reduction factor showing lower exposure of personnel:

$$K_c = D_n/D_k; \qquad (3)$$

$$K_c = MD_n/MD_k; \qquad (4)$$

where D_n and D_k are the absorbed or equivalent dose before and after decontamination;
MD_n and MD_k the dose rate before and after decontamination rad/hr (2-4);

$$K_c = D_n/PDD = P\,\tau\,/\,PDD \qquad (5)$$

where D_n and D_k – exposure doses before and after decontamination;
P – dose rate;
τ - exposure time;
PDD – maximum permissible dose.

The relationship between these values is given in Table 1.

Table 1. Relation between characteristics of decontamination effectiveness

Df	***1***	***10***	***20***	***50***	***100***	***1000***	***10000***
$\boldsymbol{D_d}$	0	1	1,3	1,7	2,0	3	4
α_d, %	100	10	5	2	1	0,1	0,01
β_d, %	0	90	95	98	99	99,9	99,99

However, the main criterion of decontamination is the extent to which contaminants are removed, i.e. the radiation safety of equipment in accordance with international or national safety requirements.

The choice of decontamination techniques depends on the specific equipment design, the character of contaminated material, the level and conditions of contamination, the decontamination level desired, etc. An optimum decontamination technique selected provides the maximum cost-effective removal of contaminants with a minimum of radwaste and personnel exposure. Decontamination consists of desorbing radioactive contaminants or stripping oxide films and deposits from the surface. Its mechanism is closely related to the nature of the sorption and sorbing surfaces [13].

Adhesion and adsorption underlie radioactive contamination processes affected by the nature of the surface and contaminants being in contact, chemical properties and form of radionuclides, sorption conditions, etc.

3. COMMONLY USED DECONTAMINATION METHODS

3.1. Decontamination with using of chemical solutions

Based on the dissolution behavior of Fe and Cr oxides, one can select decontamination solutions for removing loose corrosion deposits. [14-16]. The composition of dense oxide films, especially on stainless steel, is often more complicated. That makes the optimum choice more difficult. Sometimes two or three sequential implementations of solutions are successively used, each for a certain oxide component. Thus, organic acids are useful for dissolving hematite and magnetite, oxidant-containing alkali for Cr oxides, acidic fluorine -containing solutions for Fe, Cr, Ni and Ti oxides, reductants for higher Fe oxides.

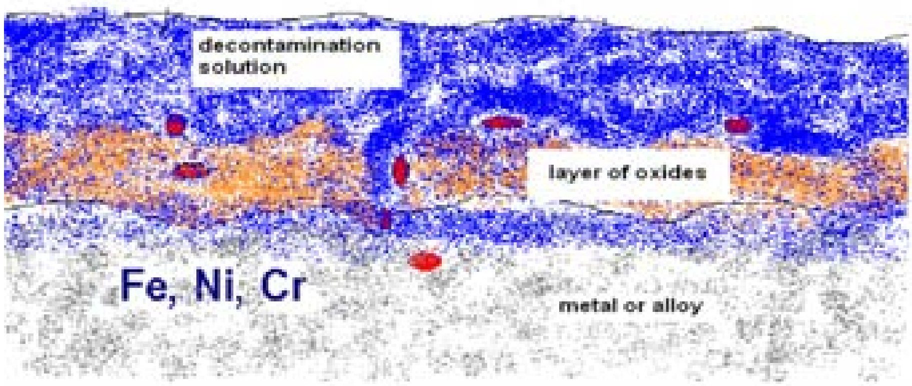

Fig. 1. Schematic view of decontamination of metal or alloy with chemical solutions.

Radioactive contaminants can penetrate deeply into metal structures, especially structural alloys (Fig. 1). Therefore decontamination often calls for the removal of surface layers. The dissolution of metals and oxides involves the sorption of water, hydroxyl ions and anions on the surface, the formation and desorption of sorbed metal complexes. Ionic Fe and HSO_4^- dissolve hematite to a greater extent than Fe metal. This is of great importance because these formulations can remove corrosion deposits, leaving metal intact. Chemical compositions and other parameters of decontamination solutions are summarized in Tables 2 and 3.

Table 2. Water solutions of chemicals for decontamination

№	Composition	Decontamination		DF	Speed of corrosion g/m^2*h		Main purpose
		t^{o}C	τ, h		SS	CS	
1	2	3	4	5	6	7	8
1	10-13% H_3PO_4 inhibited	60-65	0,5-1	4-10	0,14-2	16-26	Equipment from CS*
2	1-9% $H_2C_2O_4$ inhibited	85-95	1-4	1,3-3	0,85	6-20	Equipment from SS*
3	1,3-10% $(NH_4)_2HC_6H_5O_7$ inhibited	85-95	1-4	2-12	0,03	2-10	SS
4	2,5%$H_2C_2O_4$ + 5%(NH4)2HCit + 2%$Fe(NO_3)_3$ + 0,1% thiourea	85	1-4	3-4	-	2-10	For the demountable equipment, removal of friable sediments
5	3%H_2SO_4 + 1%$H_2C_2O_4$ + 0,1% phenylthiourea	45-70	0,5-1	-	0,02	24	Equipment from SS and Al alloys
6	1,5%$NaHSO_4$ inhibited	30-80	1-4	2-7	0,02	14-42	Equipment from CS
7	4,5-9% NH_2SO_3H inhibited	45-80	1-4	1,8-4	0,02	12.22	Equipment from CS
8	2%NH_2SO_3H + 0,5%NaF + 1,9%NaCl + DETERGENT (pH = 1,5)	95	0,5	Up to 10	1,25	18,7	For removing of corrosion sediments from the SS
9	1,5%HNO_3 + 0,2-0,5% $H_2C_2O_4$ + 0,2% NaF	50-100	-	4,3	0,25	-	For removing of slow soluble oxides from the surface of SS
10	0,06-3% EDTA (pH = 9-10)	60	1-3	-	0,007	10	For removal of friable sediments
11	10% $(NH_4)_2$Hcit + 0,4% Trilon B + 0,45% phenylthiourea	85	1-4	4,2-8	-	-	For sediments
12	2,5% NH_2SO_3H + 0,35% Trilon B + 0,5% N_2H_4 + 0,1% urotropin	-	-	1,7-7	-	-	SS
13	0,23% $H_2C_2O_4$+ 3,2% $Na_2C_2O_4$ + 1,5% H_2O_2 + 0,5% peracetic	80	1-4	2,2-7,5	-	-	U and fuel

* SS-stainless steel; CS- carbon steel;

Table 3. DF for SS using two and three reagents and fuel cycles

№	Composition	Decontamination			DF
		t,° C	τ, h	Number of cycles	
1	2	3	4	5	6
1	I - 0,3-1% $KMnO_4$ + 3-5% NaOH (KOH) II - 3% $H_2C_2O_4$	70-95	1-2	1	8-17
2	I - 3-5% NaOH (KOH) + 0,3-1% $KMnO_4$ II– 3% $H_2C_2O_4$	90	4-7	3	Up to 50
3	I - 1% NaOH (KOH) + 0,1% $KMnO_4$ II - 3% $H_2C_2O_4$	95 90	0,15 1	3-5 1	3,3-4,5 5
4	I - 5% NaOH (KOH) + 0,3-1% $KMnO_4$ II - 3% $H_2C_2O_4$ III - 0,1-5% HNO_3	95 90-100	0,15 1-2	3-5 -	5-20 8-27
5	I - 5% NaOH (KOH) + 0,3-1% $KMnO_4$ II - 0,5-1% $H_2C_2O_4$ III - 0,5-1% hexametaphosphate Na	90-100	0,35	3-5	6-9
6	I - 5% NaOH (KOH) + 0,3-1% $KMnO_4$ II - 0,1%H_2O_2	90-95	0,25-0,5	-	6-15
7	I - 3-5% HNO_3 + 0,3-1% $KMnO_4$ II - 1% $H_2C_2O_4$	90-100	0,15-0,35	3-5	23-32
8	I - 10% NaOH (KOH) + 1% $KMnO_4$ II - 10% $(NH_4)_2HC_6H_5O_7$ (inhibited)	85-95	1-4	-	~103
9	I - 1% NaOH (KOH) + 0,1% $KMnO_4$ II - 1% $(NH_4)_2HC_6H_5O_7$ (inhibited)	25	24	-	4,5
10	I - 10% NaOH (KOH) + 1% $KMnO_4$ II - 10% $(NH_4)_2$HCit + Trilon B + 0,45% phenylthiourea	90-105	1-4	-	3-56
11	I - 1% NaOH (KOH) + 0,1% $KMnO_4$ II - 0,25% $H_2C_2O_4$ III- 1% H_3Cit	90-95 95	2-6 3-9	-	40 50-120
12	I - 3-5% NaOH (KOH) + 0,3-1% $KMnO_4$ II - 0,25% $H_2C_2O_4$ III - 1% H_3Cit	70-80	1	-	10
13	I - 10% NaOH (KOH) + 1% $KMnO_4$ II - 4,5-9% NH_2SO_3H (inhibited)	70-105 70	1-2 12	- -	3-10 5-28
14	I - 10% NaOH (KOH) + 1% $KMnO_4$ II - 2,5% $H_2C_2O_4$ + 5% $(NH_4)_2$HCit + 0,1% thiourea	80-110	2	2	5,3-23
15	I - 3% NaOH (KOH) + 0,3% $KMnO_4$ II - 1,5% $NaHSO_4$ (inhibited)	80-105	1-4	-	1,5-2,0

The main advantage of chemical decontamination is high DF.

The main disadvantage of chemical decontamination methods with aggressive solutions is great volume of secondary liquid radioactive wastes with complicated salt composition, high acid or alkali content.

3.2. Water/vapor jet decontamination

Water/vapor jet decontamination uses the kinetic energy of water flow for removing of contamination, and high temperature for acceleration of chemical reactions between decontamination solutions and contaminated surface. The joint action of the kinetic energy of the jet and high temperature is more effective than solutions of chemicals or detergents. Amounts of secondary radwaste considerably decrease. The temperature of jet stream at 200 mm from nozzle is 50-100^{o}C.

A hydro-jet with chemicals is applied for removing heavy contamination with deep penetration of radionuclides into the stainless steel or alloys. A hydro-jet with abrasives is very effective for decontamination of external surfaces of buildings and constructions with deep penetration of contaminants into concrete or other constructions.

Hot alkali solutions of surfactants are useful for removing grease that traps contamination. There is a variety of steam mixtures for jetting. The choice depends on many factors, especially if there is a need for minimal corrosion of structural materials.

The main advantage is reducing secondary liquid radioactive wastes; the main disadvantage of water/vapor jet decontamination method is intensive formation of radioactive aerosols.

3.3. Application of ultrasound for decontamination

Ultrasound is used for an intensification of the decontamination process. It may allow for reducing time of decontamination by a factor of 2-3 and more: recommended decontamination of the equipment is with a frequency range of 18-40 kHz.

Application of baths for processing of small-sized or “detailed” components and tools, on which surfaces there are almost insoluble adherent or dispersive/dusting radioactive contaminants, is the most effective.

3.4. Foaming

Applications of foam serve to reduce the reagent consumption and amount of secondary liquid waste. This technique is used for large tanks, canyons, conveying passages, pipelines and vent lines. Foam is generated by an intensive aeration with compressed air or gas. The foam-induced solution (~ 3 mm thick) wets the surface well [17-20]. The surface-solution system can be heated for acceleration of chemical processes by warm air or other foam-forming gas.

A variety of ionic and non-ionic surfactants are applicable to various contamination types. Radioactive cations are principally responsible for contamination and are subject to the use of anionic surfactants. A variety of ionic and non-ionic surfactants are applicable to various contamination types. Radioactive cations are principally responsible for contamination and are subject to the use of anionic surfactants.

Advantages:

- small amount of chemicals and secondary wastes,
- possibilities to decontaminate equipment and constructions with complicated profiles.

Disadvantages:

- transportability of foam is insignificant.
- limited life -time of foam. After destruction of foam it is possible to have secondary contamination of a surface with radionuclides.

3.5. Mechanical decontamination techniques

The surface of buildings and constructions are often decontaminated by mechanical methods such as hydraulic blasting and hydraulic abrasive blasting. With hydraulic blasting, the surface is cleaned by a water jet delivered at a pressure of 80-100 kg/cm^2 and a flow rate of 1-2 m^3/hr. As a function of the material morphology, the hydraulic blasting output typically varies from 1.5m^3/hr (for asphalt) to 3.0 m^3/hr (for unpolished facing stone). Hydraulic blasting has shown the removal of contaminants by a factor of 1.7-6, depending on the material (1.7 for concrete, 6 for red brick).

It is possible to subdivide mechanical methods into two main variations. Hydraulic abrasive blasting involves the combination of a high-pressure water jet (80-100 kg/cm^2) and abrasive material (sand for example) added to the jet. This is much more effective than hydralic blasting alone. The treatment rate is 4-6.7 m^2/hr at an abrasive consumption of 50-70 kg/hr. Hydraulic abrasive blasting is of particular utility for removing contaminants penetrated deeply in the interior of the building outer surfaces.

The second approach involves mechanical crushing of surface using shock-and-vibration or scabbling. For tests at MosSIA “Radon” our team used a PENTEK VAC-PAC dustless system. It includes a vacuum system for removing contaminated debris with dust equipped HEPA filters, a system of safe collection of concrete debris into 55 gallon drum and a set of specially designed tools for scabbing and abrasive treatment of surface-in-air media. The capacity for each tool varies with conditions of operation/operator skill (Table 4).

Table 4. Depth of treatment of concrete surfaces

№	Tool	Dept of treatment, mm
1	Roto-Peen for decontamination of flat surfaces	1
2	Corner-Cutter for decontamination of corners and profiles	1
3	Squirrel II for decontamination of cracks in concrete	5
4	Squirrel I for decontamination of local contamination	2
5	Squirrel III for decontamination of floors	3

Table 5 summarizes the main results of our analysis of available decontamination techniques which may be recommended for actual decontamination works.

Table 5. Summary list of decontamination techniques

##	Decontamination Method	Application	Advantages	Disadvantages
1	**Chemical decontamination**			
1a	1 reagent	CS, SS, plastic	High DF	Large volume of RW,
1b	2 and more reagents	CS, SS, plastic	High DF, applied for deep penetration of contaminants	Preferable for fragmented equipment

##	Decontamination Method	Application	Advantages	Disadvantages
2	**Water/steam jet**			
2a	With surfactants	Non-fixed contamination onto metal, ceramic and plastic, painted surface	Simple application	High aerosol formation Only for non-fixed contamination, foam formation
2b	With chemicals	Decontamination from heavy contamination	Possibility of decontamination of wide list of subjects	High aerosol formation Need high temperature of application (60-120^{0}C)
2c	With abrasives	Concrete, brick, painted/corrosive CS and SS		
3	**Foaming**	Non-fixed contamination onto metal, ceramic and plastic, painted surface	Low RW formation	Transport ability of foam is insignificant. Life -time of foam is limited, and after destruction of foam probably secondary contamination of a surface with radionuclides
4	**Ultrasonic**	Non-fixed contamination onto metal and plastic	Low RW formation	Small capacity
5	**Electrochemical decontamination**			
5a	In bath	CS, SS, fragmented equipment		Formation of aggressive liquid RW
5d	External electrode	CS, SS, plastics	Low RW formation, wide application possibilities includes non-fragmented equipment, floors, walls	
6	**Strippable coatings**	All kinds of surfaces except equipment with complicated profile	Low solid RW formation	Low mechanical strange of polymeric films

##	Decontamination Method	Application	Advantages	Disadvantages
7	**Thermo decontamination**	Concrete, brick, painted/corrosive CS and SS	Low RW formation	method may apply only for horizontal surfaces of concrete, bricks and corrosive/painted surface of metal
8	**Mechanical decontamination**			
8a	Abrasives	painted/corrosive CS and SS, concrete, bricks		Using of special tools and adaptive ones
8b	Scrabblers	Concrete, brick, painted/corrosive CS and SS, painted surfaces		
9	**Ice blasting**	Equipment with complicated profile, CS, SS, painted surfaces	Low RW formation	High aerosol formation

4. DEMONSTRATION OF ADVANCED TECHNIQUES

In the framework of this CRP, two decontamination methods were chosen and tested:

- Electrochemical decontamination with external electrode;
- Polymeric strippable coatings.

4.1. Electrochemical decontamination with external electrodes

Decontamination can be improved using electrolytic processes. It is possible to save time and reagents, reduce waste, mechanize and control the treatment and conduct it at low temperatures without sacrificing performance. The most commonly used technique is anodic dissolution producing readily soluble substances. The anodic dissolution of metal in the form of simple hydrated ions or complex ions is the opposite of cathodic process in many respects. The anodic dissolution reaction for simple hydrated ions formed can be written as:

$$M + xH_2O = M^{z+}\ xH_2O + ze^- \qquad (6)$$

or

$$M + xA^- + yH_2O = MA_x^{z-x}\ .\ yH_2O + ze^- \qquad (7)$$

for complex ions.

Electrochemical decontamination under cathodic polarization is used for removing loose and non-fixed contaminants. Oxide films are partially or completely reduced on the metal, with hydrogen evolved intensifying their destruction. The hydrogen released is able to detach loose radionuclides from the metal surface. As a rule, insoluble contaminants are removed in this process of decontamination with high DF. However the cathodic dissolution sometimes involves the risk of re-contamination, especially by radionuclides with positive electrode potentials.

The method of ion-exchange electrochemical decontamination is a combination of a bipolar electrochemical process and ion-exchange sorption of radionuclides on fibrous materials. It is characterized by a higher DF, a low volume of secondary liquid radwastes and a high productivity [21-22].

For carrying out electrochemical decontamination of large metal surfaces, handy electrodes were developed and tested. These handy tools consisted of an external, simple electrode with ion exchange material and electrode - roller subdivided into 2 parallel electrodes with special external felt seal of a new design. The felt seal served for constant feeding of the process with electrolyte solution. Sorbents with a capacity of 2-3 mg-eq/g were used for the tests. (Figs. 2 and 3).

Fig. 2. Test of roller on vertical wall of glovebox.

Fig. 3. View of simple electrode.

Tests of the handy electrodes were carried out in the laboratories. Samples from stainless steel 12Ch18N10T with size 1.0 x 1.2 m contaminated with UO_2 were used for tests. The consumption of electrolyte was 50-70 ml / m^2. Results are presented in Table 6 and 7.

Table 6. Electrochemical decontamination of stainless steel (Roll electrode)

Contamination, Bq/sm^2		Decontamination Factor
initial	After treatment	
85,0	0,1	850
69,0	0,1	690
77,0	0,1	770
56,8	0,1	568
63,6	0,1	636

Table 7. Electrochemical ion-exchange decontamination of stainless steel (flat electrode)

Contamination, Bq/sm^2		Decontamination Factor
initial	After treatment	
93.2	0,1	932
61.3	0,1	613
88.5	0,1	885
48.9	0,1	489
56.8	0,1	568

Results of pilot tests showed the high efficiency of the handy tools in the case of stainless steel contaminated with UO_2.

The main advantage of this technique and tools is the possibility to decontaminate large metal and plastic surfaces without fragmenting them.

4.2. Polymeric Strippable Coatings

Polymeric strippable coatings are using for decontamination and interim localization of contaminants on the inner surface of buildings and on the outer surface of equipment for prevention of distribution of contamination. Formulations for these coatings generally share a polymeric basis and are doped with surfactants, sorbents, chargers, peptizers for adjustment of strength, elasticity, adhesion, etc. Aqueous acryl and vinyl solutions, aqueous emulsions of polymers (polyvinyl acetate), aqueous dispersions of rubber, water-based double and triple copolymers are in common practice as coating formers. The use of organic solvent-based polymers (methylethylketone, toluene, etc.) is limited because of their fire-, explosion- and toxicity hazard. Peptizers for water-based polymers are high molecular alcohols (glycerin), organics (tributylphosphate), and mineral acids (phosphoric acid). Surfactants are the nitrates of synthetic fatty acids: sulfonol, formulations with trade mark OP-10. For tests and actual application, two coating types were used differing in so-called active additions providing protective or decontaminating properties.

Effective decontaminating coatings call for aggressive ingredients like mineral and organic acids, oxidants and alkaline metal hydroxides. [21-23]. They favor the inclusion of radionuclides (surface contamination) into the polymeric coat structure or the destruction of the most contaminated surface layer (deep contamination) followed by the removal of contaminants and interaction products. Thus contaminants penetrating 1-50 μm can be removed using a number of special additives including aggressive ones for different kinds of surface and different nature of contaminants.

Poorly-soluble radioactive dust can be removed by the adhesion technique with polymeric films. For this purpose the dusty surface is covered with a coating-forming formulation containing surfactants and anti-adhesion agents to stabilize the dusty contaminants by inclusion into the coating formed. The process of decontamination is completed by stripping and removal of coatings.

Out of a variety of coat-forming agents tested, environmentally- and fire- safe acrylic and vinyl polymers have been selected. These involve surfactants, glycerin, filling and complexing sorbents. Aggressive decontaminating compositions can have mineral or organic acids and various oxidants added. It takes 4-8 hours for a polymeric coat ~100 μm thick to be produced at 20^0C and a humidity of 70%, the stock consumption being 0.25-0.5 l/m^2. The coat remains protective for more than 100 days. The coat adhesion to surfaces of flexible PVC, carbon steel, stainless steel or paintwork is 1-2 g/cm^2.

DF in laboratory tests and in actual practice:	carbon steel	– 10-20
	stainless steel	– 20-50
	concrete, bricks	– 5-7

View of strippable coating onto red brick, ceramic and metal surfaces (Figs. 4-8).

Fig. 4. Brick surface.

Fig. 5. Ceramic surface.

Fig. 6. Metal surface.

Figs. 7 and 8. Strippable coatings with metal oxides.

For the tests PVA-based strippable coatings were selected and doped with aggressive agents: decontamination coatings (DC) and pickling-decontamination coatings (PDC). More than 160 real and artificially contaminated specimens were tested. Tables 8, 9 and 10 present the results of decontamination by strippable polymeric coatings currently employed.

Table 8. Decontamination efficiency for real stainless steel specimens contaminated with ^{137}Cs

Formulation	*Initial activity, Bq*	*Residual activity, Bq*
DP-1	$1*10^3 \pm 1*10^2$	background*
DP-2	$1*10^3 \pm 1*10^2$	background
PD-1	$1.1*10^3 \pm 1*10^2$	background
PD-2	$1.2*10^3 \pm 1*10^2$	background

* an average of 5 parallel experiments

Table 9. Decontamination efficiency for actual ^{90}Sr -contaminated stainless steel specimens

Formulation	*Initial β activity, Bq*	*Residual β activity, Bq*	*Residual β activity, Bq*	*Residual β activity, Bq*
		1st cycle	*2nd cycle*	*3rd cycle*
DP	$1.9*10^3$	$3.3*10^2$	$2.5*10^2$	$0.15*10^2$
TDP-3	$1.9*10^3$	$2.6*10^2$	$1.0*10^2$	background
TDP-4	$1.2*10^3$	$1.5*10^2$	$0.8*10^2$	background

*an average of 5 parallel experiments

As evidenced by our study, the removal of artificial contamination was highly effective. The residual activity of ^{137}Cs, and ^{90}Sr was as low as background.

The starting decontamination formulation consumption required to produce a coating is 0.26-0.5 kg/m^2. A coating thickness of 80-120 μm is optimum. It takes from 2.0 to 6.0 hours at 22±2^{0}C and a relative ambient humidity of 75% for a coating to be developed. At this consumption the decontamination produces 100-300 g/m^2 of solid radioactive waste.

Table 10. Decontamination of samples of different nature

№	**Sample**	**β-activity, Bq/cm^2**		**DF**
		initial	final	
1	Stainless steel	500	Background *	--
		500	Background *	--
		500	Background *	--
		10000	Background *	--
		20000	20	1000
2		500	Background *	--
		500	Background *	--
		600	Background *	--
		8000	Background *	--
		15000	15	1000
1	PVC-Plastic	500	Background *	--
		700	Background *	--
2		500	Background *	--
		800	Background *	--

* Background is 20 Bq/ cm^2

Results show that both polymeric compositions remove radioactive contaminations from the stainless steel and PVC-plastic with high efficiency.

4.3. Summary

Results of tests in the laboratory and real experience have shown high efficiency of electrochemical decontamination with external electrode and polymeric strippable coatings. Both techniques have good perspectives for wide application as simple methods with high DF and small amount of secondary radioactive wastes.

5. PREPARATION OF THE ALGORITHM FOR DECISION-MAKING PROCESS

In accordance with [24] the decision-making process and process of preparation for decontamination should fulfill the following basic criteria:

- be consistent with rules of logic,
- be transparent,
- take account of views of all stakeholders,
- take account of all factors affecting the decision making process,
- give balanced consideration for all possible options for action,
- provide unambiguous advice.

Special attention should be to ensure the decontamination plan and works are in accordance with international and national safety standards and regulations.

Safety standards and regulations in the Russian Federation
There are 3 main regulations in the Russian Federation in the field of D&D [25-27]:

(1) Rules of maintenance for radiation safety (OSPORB-99), Sanitary Rules 2.6.1.799-99
(2) Norms of radiation safety, (NRB-99) 1999
(3) Sanitary rules of the management of radioactive waste (SPORO-2002)

5.1 Rules of maintenance of radiation safety (OSPORB-99), Sanitary Rules 2.6.1.799-99

Terms and definitions

The basic Sanitary Rules of maintenance of radiation safety establish requirements on protection of people against radiation influence under all conditions of an irradiation from sources.

Rules are obligatory for design, construction, operation, reconstruction, and decommissioning of “radiation objects” (facilities involving radioactivitiy). Rules offer that the decision on prolongation of operation or decommissioning of a “radiation object”, and also the choice of strategy are accepted only after inspection of the radiation situation and technical condition of systems and equipment, building constructions and territory and of the facility.

The decommissioning project plan should contain:

- Preparation of the equipment necessary for dismantling works;
- Methods and means of decontamination of equipment which should be fragmented;
- Radioactive waste management.

No restrictions are applied on use of solid materials, raw materials and products with specific activity less than 0,3 kBq/kg. Materials, contaminated to levels higher than the values below should be managed as radioactive wastes (Table 11).

The raw material, materials and products with specific beta-activity from 0,3 up to 100 kBq/kg, or with specific alpha - activity from 0,3 up to 10 kBq/kg, or with the contents of TRU from 0,3 up to 1,0 kBq/kg can be used, subject to limits on the basis of the conclusion of the state sanitary-andepidemiologic authority on the specific application. These materials are subject to obligatory radiation control.

Table 11. Classification of liquid and solid radioactive waste

Category of radioactive waste	Specific activity, kBq/kg		
	Beta-radionuclides	Alpha-bearing radionuclides (except TRU)	TRU
Low-level	Less than 10^3	less than 10^2	less than 10^1
Medium-level	10^3- 10^7	10^2 - 10^6	10^1 - 10^5
High-level	More than10^7	More than 10^6	More than 10^5

5.2. Norms of radiation safety, (NRB-99) 1999

Norms of radiation safety NRB-99 are applied for human safety in all conditions of influence of ionizing irradiation. The effective dose of an irradiation of natural sources of the workers who are not concerning to a category the personnel, should not exceed 5 mSv/year under production conditions (all trades and manufacturing).

The basic limits of dose irradiation from technological sources for various categories of irradiated persons (Table 12) are established in NRB-99.

Table 12. The basic limits of doses

Dose	Limits of doses	
	Personnel, group A	**Population**
Effective dose	20 mSv/year, average for any consecutive 5 years, but no more than 50 mSv/year	1 mSv/year, average for any consecutive 5 years, but no more than 5 mSv/year
Equivalent dose per year in a crystalline lens of an eye	150 mSv	15 mSv
To skin	500 mSv	50 mSv
Brushes and foots	500 mSv	50 mSv

5.3. Sanitary rules of the management with radioactive waste (SPORO-2002)

Sanitary rules establish requirements for radiation safety of personnel and population for all radioactive waste management.

Radioactive wastes are subdivided as liquid, solid and gaseous. For an unknown radionuclide composition, solid wastes will be considered as radioactive if their specific activity is more than:

- 100 kBq/kg - for beta- radionuclides ;
- 10 kBq/kg - for alpha–bearing radionuclides;
- 1 kBq/kg - for TRU.

6. DEVELOPMENT OF DECONTAMINATION PLAN

Based on experience of Mos SIA “Radon” D&D projects there are a few milestones in the preparation and decontamination process to take note of:

(A) At the Pre-decontamination stage, define:
- Initial data treatment – records, archives etc.
- Monitoring – mapping, sampling, spectrometry.

(B) Preparation of:
- Lists of contaminated equipment (types, amount, levels, size);
- Lists of contaminated rooms (levels, size, paint, deep of penetration of contamination).

(C) Establish knowledge of decontamination criteria and sanitary limits.

(D) Setup decontamination criteria, safety requirements, interact with regulatory body, local administration and stakeholder: Knowledge of future plans (restricted or unrestricted usage, "green field"...)

(E) Development of schedule and selection of decontamination techniques:
- Methods, equipment which will apply during decontamination process;
- Estimation and analysis of doses;
- Planning of man-hours;
- Accident plan;
- Waste management strategy;
- Certification and preparation of sanitary-epidemiology conclusion;
- Project management;
- Team creation;
- Interaction with stakeholders, regulators, local administration, service team, contractors;
- Development of technical base (purchasing/rent equipment, tools, machines, materials etc);
- Choosing, purchasing or renting of equipment;
- Management of project in progress;
- Main principles of inspection and characterization after decontamination.

(F) Decontamination activity:
- Preparation of building to decontamination operation, subdivide to two zones according to standards for “clear” and “contaminated” zones;
- Preparation and equipping of sanitary posts, counting systems, area for waste collection, preliminary localization of contamination using protective coating. Preparation of pathway for radioactive waste transportation on site to temporary waste storage area;
- Preparation of tools and equipment for safe dismantling of contaminated equipment and constructions;
- Decontamination of equipment and constructions;
- Dismantling or cutting of contaminated equipment and constructions;
- Removing of radioactive wastes, accumulation of it at a special site, transportation to waste treatment/storage facility.

(G) Demolishing (if necessary):
In the case of liquidation of a facility, the main and supporting buildings will be demolished carefully. Attention should be given to the application of special machines equipped with dust suppression or dust evacuation systems.

(H) Waste management:
Wastes (solid and liquid) collected as a result of decontamination activity must be pre-treated, packaged in transporting containers and shipped to a special facility for treatment, isolation and safe storage. Variant of waste treatment on site should be chosen based on requirements of national standards.

(I) Site remediation and inspection:
For site remediation, use techniques in accordance with the anticipated future status of the site, e.g. industrial or urban.

(J) Inspection and finalization of works.

(K) Reports for regulatory body, local administration and stakeholder.

6.1. Development of Decontamination Plan (based on experience of MosSIA "Radon")

The Scientific and Industrial Association "Radon" (MosSIA "Radon") is a State Unitarian Enterprise of the Government of Moscow responsible for decontamination, transportation, treatment and long-term storage of non-fuel cycle radioactive wastes arising in the Central Region of Russia. This region played a very important role in the development of the Soviet nuclear industry and nuclear weapons. A lot of historical research institutions and industrial facilities that developed nuclear technologies are located in this area with a population of more than 50 million citizens. There are more than 2000 facilities which delivered radioactive wastes to Mos SIA "Radon" at the present time.

As a result of the transformation of the economic situation in Russia at the end of the 20th century, these sites have financial problems, partially changed owners or even loss of control. Nuclear research and industrial facilities formerly located in suburban areas are now located in the city limits with high population density at present as a result of urbanization. The problems of safe decontamination, decommissioning and remediation of contaminated sites especially "legacy sites" is an urgent necessity.

A special division for decontamination service and emergency works was established in 1994 in Mos SIA "Radon". At present the Center of Radioactive Waste Transportation and Emergency Service of Mos SIA "Radon" provides full service in site decontamination, decommissioning of research and industrial facilities contaminated as a result of former nuclear activity. Other duties of the Center include developing new techniques and new equipment, modifying well-known ones, and creating approaches for decontamination and full scale remediation.

Experience of the Center since 1994 includes:

- More than 650 historical sites decontaminated;
- More than 23000 cubic meters of radwastes transported for treatment and storage;
- More than 800 emergency calls;
- More than 10 industrial/research sites decommissioned.

To take into account the high population in the area of responsibility, every decontamination project is unique and needs special management, research, radiological survey, engineering, monitoring etc. One of the main limitations is compliance with Russian regulations and safety standards.

Development of a decontamination plan is a multistage and complicated procedure which should join technical and safety aspects of D&D works. The following factors have been identified as very important in the preparation of D&D projects in accordance with national safety standards and MosSIA “Radon” practices.

Initial data collection

For all objects which were commissioned 40-50 years ago, there is no trustworthy information about:

- design of buildings and modifications;
- communications;
- equipment;
- contamination and radionuclide composition;
- location of accidental contamination.

Experience of MosSIA “Radon” has shown that each D&D site is unique; therefore for any project, it is necessary to begin from a search of all kinds of information which may be available. Results of initial data collection and treatment may include:

- historical records, interview with former operators;
- design of the project;
- construction records and records about modification in design of the object;
- radiometric and spectrometry data, engineering survey;
- information about technological processes;
- period of operation of the facility;
- operational accidents;
- data and shutdown procedure;
- list of equipment to be decontaminated (types, amount, levels, size);
- list of contaminated rooms (levels, size, paint, depth of penetration of contamination);
- knowledge of future plans (restricted or unrestricted usage, "green field");
- decontamination criteria and limits in accordance with safety requirements.

Complex of engineering - radiological inspection of object (KIRO):

Main goals: objective estimation of a condition and properties, structure and contamination; estimation of assumed character of distribution and fixing of contamination.

Tasks performed by KIRO:

- Carry out engineering and radiological inspection of constructions to determine its mechanical condition;
- Estimate correspondence of real designs and constructions with the initial design documentation;
- Determine deviations from the initial design documentation; (inspection of presence of defects and damage of the building construction resulting in decrease of their bearing ability and interfering the further normal use of the building;
- Fix the condition of the process equipment and systems (system of ventilation, system of plumbing and sanitary, system of a heat supply, system of electro supply and illumination, the monitoring system of radiation safety etc.);
- On the basis of detailed inspection of radiation conditions of the object, a “map” (cartogram) of contamination is made, radionuclide composition determined, levels of contamination and a rough estimation of the radwaste amount calculated.

Development and certification of the decontamination plan

Based on the decommissioning concept, safety standards and results of radiation and engineering, the survey decontamination plan should include a list of techniques, methods, tools and equipment, operations and materials etc. which are needed for safe and successful operation.

The typical creation of a decontamination plan includes:

- Detailed characterization of site, radiological survey, sampling (Figs. 9 and 10);
- Mapping of contamination;
- Result of site characterization with detailed results of engineering and radiation inspection;
- The concept of normalization of radiation conditions, decontamination criteria and limits ;
- Requirements for work and the order of operations (subdividing of buildings to "contaminated" and "clean" zones, equipment of sanitary posts, preparation of necessary communications and equipment, a marking of routes of moving and places for temporary storage of radioactive wastes);
- Description of technologies, ordering specifications, materials, special equipment, tools for work;
- Main goals, criteria and limits of decontamination work; planning of man-hours; accident plan; waste management strategy;
- Radiation control and maintenance of radiation safety;
- Safety requirements;
- Detailed plan of control measurements after completing of decontamination works.

Fig. 9 Radiological survey in old radiochemical laboratory with known radionuclide composition.

Fig. 10 Radiological survey with application of gamma-, beta- spectrometers and dosimeters in the case of unknown radionuclide composition.

One of the basic and most complicated points of the plan is the choice of optimal techniques and equipment for decontamination. The choice depends on character of contamination, levels, scales, the nature and structure of contaminated material (see Tables 2,3,5), final goals of decontamination, criteria and limits of decontamination works, design of building, waste management rules, and the budget.

Decontamination criteria should fit with requirements of national safety requirements, for example in the Russian Federation for industrial site, it is possible to use the following criteria and recommendations for handling of industrial (non radioactive) wastes (Tables 13 and 14):

Table 13. Criteria of completeness of decontamination work (For site "X")

№	Name	Unit	Type of contamination	
			territory	rooms
1	Gamma dose rate	mR/h (mSv/h)	60 (0,54)	60 (0,54)
2	Fixed surface beta contamination	Beta-counts/min*sm^2	non	50
3	Non-fixed surface beta contamination	Beta-counts/min*sm^2	non	non
4	Fixed surface alpha contamination	Alpha-counts/min*sm^2	non	5
5	Non-fixed surface alpha contamination	Alpha-counts/min*sm^2	non	non

Table 14. Recommendations on use of construction wastes and ground after decontamination which contain radionuclides less than safety limits (for site "X")

№	Specific activity, Bq/kg	Utilization
1	Less than 370	No restrictions
2	370 - 1000	Industrial waste dumping site

The key section is a description of decontamination approach in detail. It should include all information about technical operations, sequence, tools, chemicals etc. A special part is the method of waste management at this stage. An example of a technical chart for the preparation stage of decontamination (for site "X" is in Table 15).

Table 15. Example of approach to decontamination work in "factory sections #14"

Task	Operation	Instructions	Detail description
Carrying out of primary decontamination with the purpose of removal or fixation of non-fixed contamination, preparation of workplaces for decontamination	Removal or fixation of non-fixed contamination from/on equipment, walls and floors	Remove plastic cover with defects from floor in room #s 5, 10, 11. Fragment it. Rectangular fragments in length of 2-4 m, further reuse, if necessary, as additional protection. To remove radioactive dust using industrial vacuum cleaner "MAKITA". Fixation of non-fixed contamination on walls, floors and equipment with polymeric coats or acrylic paint.	Fixation of floor surface. Carry out using paint with deep penetration ("Optimist") with creation of two-layer cover with interim drying time of 30 min. Fixation of contaminants on walls and equipment carried out using TDP-3 PVA-based polymeric strippable coating composition using rolls or brushes for the case of small or non-linear area of contamination. Consumption is 0.25-0.5 l/m^2

For the decontamination plan, it is necessary to agree with customers, local administration and regional division of civil defense of the Ministry of Emergency, to take their advice and receive the "sanitary-epidemiologic conclusion" (certificate).

Preparation of site for decontamination

Preparation for work of personnel of group A include: subdividing building or site to zones ("Clean" and "contaminated"), equipping of a sanitary post on the border of zones, dosimetry, routes of moving of the personnel, materials, equipment and radioactive wastes (Figs. 11, 12).

Fig. 11. Example of temporary sanitary dosimetric post in the room.

Fig. 12. Dosimetric post outside building.

Special places for packaging and temporary storage of radioactive wastes should be equipped. Preparation of workplaces for operators and training of all personnel are required.

Purchase, delivery of unique or special equipment, and provide materials before beginning decontamination work.

Decontamination, dismantling

Decontamination and dismantling of the large-sized equipment and building construction are made according to the decontamination plan. First of all is removing or fixation of non-fixed contamination to prevent its distribution. Before dismantling of large-size equipment and building construction all contaminated surfaces are covered by paint, strippable coats or other methods (Figs. 13-14).

Fig. 13. Treatment of contaminated concrete with dust suppressed polymers.

Fig. 14. Example of application of reinforced polymeric protective film on the stair.

Special decontamination techniques are needed for decontamination of painted, polymeric or metal surfaces. Decontamination of concrete and bricks is often completed after removing the upper layer of contaminated material. For these purposes it is desirable to use commercial tools which are able to remove up to 5 cm of concrete (Fig. 15) or alternatively special machines (Fig. 16). Main restrictions involve application of vacuum systems for removing of dust or dust suppression with special polymers.

Fig. 15. Decontamination of concrete with handy tools.

Fig. 16. PENTEC VAC-PAC System with Squirrel II decontamination tool.

The strategy of work: highly contaminated materials are removed first. As a result, doses for future operations decreased, lesser areas of contamination are more "visible" as a result of the decrease of background. Decontamination of walls is made top-down, and for horizontal surfaces from edges to the center of the contaminated area.

Collection and packing of radioactive wastes

Collection and packing of radioactive wastes is made according to the requirements of national safety standards and regulations; requirements of transporting, processing and storage facility. Usually, waste is packed into a primary pack (a polyethylene bag or extra-strong paper) at the place of arising, then delivered to the preliminary and temporary storage area, placed in transport containers and delivered to the processing facility according to the accepted decontamination plan (Figs. 17, 18). Each set of wastes is supplied with a certificate with information about type of waste (solid, liquid, combustible/non-combustible, compacted/non-compacted) radionuclide composition and total activity.

Fig. 17. Packaging of radioactive wastes on-site.

Fig. 18. Transportation of RAW to on-site temporary storage.

Inspection and assignation of object

After finishing decontamination work, radiation inspection is carried out. For each object, papers to be signed by the customer, regulatory body, local authorities and sanitary control are prepared as the basis for inspection.

6.2. Lessons Learned

The lessons learned in State Unitary Enterprise MosNPO "Radon" confirm that every contaminated site is unique and D&D plan and decontamination techniques should be modified to suit each project: there are no universal techniques and methods. At the same time it is necessary to pay attention to the presence of typical problems and solutions.

Problems of historical records

There are no correct records for at least 50 % of old (50 and more years) contaminated subjects. Each building, room, glove box or hot cell is an “unknown subject” and it needs to be subjected to a full scale monitoring of contamination and building construction. Much time is needed for development of decontamination projects taking into account searching and analyzing of information available.

Modification of commercial tools and methods for cost minimization

The modified commercial tools and equipment, or commercial chemicals applied widely for minimization of decontamination/demolishing are expensive. The modification of equipment or tools may involve to equip them with a vacuum cleaner system for evacuation and filtration of radioactive aerosols; modification with the purpose of using these tools for remote control; modification of design for decontamination needs etc. This approach may be useful for saving funds but the special equipment and tools are more effective and better designed.

Special polymeric protective/decontaminating strippable coats reinforced with cotton normally are used for protection of contaminated surfaces. For the purpose of minimizing cost, it is possible to use commercial polymeric films/commercial foam or alkyd/acrylic paints for fixation of non-fixed surface contamination (Figs. 19-21).



Figs. 19-21. Application of commercial polyethylene films, commercial polymeric foam and modified strippable coating for fixation or isolation of contaminations.

Needs of radwaste volume minimization

Fragmentation of boxes, constructive materials etc. on-site allow to minimize waste volumes to be transported for a long distance. Radwastes are packaged on-site into 200 l drums or special casks and transported to Mos SIA Radon for treatment and long-term storage.

The typical approaches to decontaminate work*:*

- First and foremost: removal of non-radioactive trash from the rooms; segregation of building to the "clean" and "dirty" zones; equip sanitary and dosimetric posts;
- Rebuilding of degraded electricity, water and exhaust ventilation systems;
- Decontamination of non-fixed contamination or its fixation with polymers or paints;
- Decontamination of walls and floors begins from the highest levels of contamination;
- Decontamination of multilevel buildings begins from the upper level and moves downwards. In the case of decontamination of communications and other rooms which are below grade, this can lead to destruction of the building; it can be done better after demolishing the building;
- The large-sized equipment should be decontaminated and fragmented as possible for minimization of volume of radioactive wastes to be stored;
- Radiometry and sorting of wastes to radioactive and non-radioactive is carried out at all stages of decontamination and demolishing works;
- Modified commercial tools are possible to use for reducing decontamination/demolishing cost;
- Water or solutions of special polymers should be used for radioactive dust suppression during demolishing of concrete or bricks, also in the case of handling radioactive wastes.

7. INTERACTION WITH OTHER CRP MEMBERS

(1) April 2005 – Technical visit, RCM in Halden, Norway (G. Rindahl).
(2) November 2006: J. Dadoumount (Belgium) visited Mos SIA "Radon" for exchange of information, technical tour to Waste Management Facility and for discussion and planning of cooperation in D&D techniques.
(3) November 2006: S. Alejandro Fabbri (Argentina) visited Mos SIA "Radon" for exchange of information, technical tour to Waste Management Facility and for discussion and planning of cooperation in D&D techniques.
(4) December 2006 – Technical visit to Belgium, Mol, information exchange with J. Dadoumount in the field of decontamination techniques which are used at the BR-3 D&D Project.
(5) December 2006: Technical visit and RCM in Keswick, United Kingdom (M. Cross).
(6) November 2007 – Technical visit to IPEN, Brazil (P.E. Oliveira Lainetti).
(7) November 2007 – Technical visit to CNEA, Argentina (S. Alejandro Fabbri).
(8) November 2007 - Technical visit to Deconta a.s, Slovak Republic (V. Daniska).
(9) December 2007 – Technical visit, RCM in Rez Czech Republic (J. Podlaha).
(10) 2008 – planning training of P. Ernesto Oliveira Lainetti (Brazil) in Mos SIA "Radon".
(11) 2008 – planning technical visit D&D professionals to Danish Decommissioning, Denmark, (K. Lauridsen).

8. CONCLUSIONS

In the framework of the CRP, the following was prepared, performed and analyzed:

(1) A short review of the scientific and technical information on the decontamination techniques which may be used in D&D activity. A summary list of more useful and modern decontamination techniques (e.g. Table 5) was prepared.

(2) Demonstration of real applied efficient decontamination methods used and tested: electrochemical decontamination with external electrode and polymeric strippable coatings. Results of tests in laboratory and real experience have shown good efficiency of electrochemical decontamination with external electrode and polymeric strippable coatings. Both techniques have good perspectives for wide application as simple methods with high decontamination factor and a small amount of secondary radioactive wastes.

(3) Based on experience of Mos SIA "Radon" D&D projects there are a few milestones in preparation and decontamination process discussed and recommended for preparation and implementation of the decontamination plan. It should include all aspects of site characterization, adherence to safety standards and regulations, description of decontamination and/or demolishing techniques, machines and tools for D&D project, and waste management.

(4) The lessons learned by the State Unitary Enterprise Mos SIA "Radon" confirm that every contaminated site is unique and D&D planning and decontamination techniques should be adjusted to each different project. There are no universal techniques and methods. At the same time it is necessary to pay attention to the presence of typical problems and solutions: problems of historical records; possibility to reduce the D&D cost in the case of modification of commercial tools and methods and radwaste volume minimization. The typical approaches to decontaminate work are summarized and may be recommended for future similar applications.

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COMPARATIVE ANALYSIS OF DECOMMISSIONING TECHNOLOGIES BASED ON MODEL CALCULATIONS AND MULTI-ATTRIBUTE ANALYSIS OF SPECIFIC DECOMMISSIONING CASES OF NUCLEAR FACILITIES

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Abstract
The decision making process for selecting scenarios and technologies for decommissioning of nuclear facilities is a multi-parametrical process. Before decision for the optimal option of decommissioning, all considered scenarios should be evaluated analytically in order that as many as possible objective data are available for selection of the optimal option using multi-attribute analysis. The paper presents the properties and examples of application of computer code OMEGA as the tool developed for these purposes. The OMEGA code is used for generation, calculation and optimisation of individual options of decommissioning in the decision making and planning phases. The main calculated parameters are cost, exposure, manpower, personnel, material and radioactivity data and the decommissioning schedule in the form of the Gantt chart. The tool implements the standardised structure of items for costing purposes and the system for on-line management of material and radioactivity flow in decommissioning process. Besides the presentation of the code, the paper deals with methods for evaluation and optimisation of decommissioning options for the purpose of selection of the optimal decommissioning option; modelling of dismantling techniques including the remote dismantling techniques; methods for evaluation of safety in decommissioning planning phase and analytical methods for optimisation of waste management scenarios including the approach for evaluation of conditional release of metals.

1. Introduction

The paper presents the results of activities performed in the company DECOM, a.s., jointly with DECONTA, a.s. in the frame of the IAEA Co-ordinated Research Project (CRP) T2.40.07 “Innovative and Adaptive Technologies in Decommissioning of Nuclear Facilities”. The name of the subproject is “Comparative analysis of decommissioning technologies based on model calculations and multi-attribute analysis of specific decommissioning cases of nuclear facilities”. The aim of the subproject is to contribute to the planned overall objectives of the co-ordinated research project especially aspects of “Conducting cost-benefit or multi-attribute analyses of specific cases of technology comparison and selection” and “Comparing innovative vs. adaptive technologies”.

The decision-making process for selecting scenarios and technologies for decommissioning of nuclear facilities is a multi-parameter process. Normally, those decommissioning scenarios are taking into consideration, the spectrum of possible procedures in decommissioning under facility, site or national specific conditions. The general procedure for selection of the optimal decommissioning scenario and the factors which should be taken into consideration are presented in several IAEA publications [1], [2]. The most important parameters in evaluation of decommissioning scenarios are the costs and safety related parameters. The general aspects having impact on the decommissioning costs are presented in IAEA publication [3] and the safety related parameters and procedures which should be taken into consideration in selection of the optimal decommissioning scenario are discussed and presented in the recent IAEA project DeSa [4].

The participants of the decision making process are the decommissioning experts who, in general, use two groups of data — subjective data and calculated data. The subjective data are the data used by the experts as estimated data or data from other projects, adapted to the evaluated facility. The calculated data are the results of calculations which are specific for the evaluated facility. The calculated data should support all decommissioning scenarios to be evaluated. The decision making process is more objective when more calculated data specific to the evaluated facility are available.

The goal of this sub-project is to present the new tool and related procedures for supporting the decision making process in decommissioning. The new tool is the computer code OMEGA, developed in company DECOM a.s. in the Slovak Republic. This computer code includes the evaluation and optimisation of decommissioning cost (as the main decommissioning parameter), safety parameters related to personnel and environment, and parameters of management of waste generated in decommissioning.

The code uses one compact calculation structure which provides all data within one calculation run. The code simulates the material and radioactivity flow in the decommissioning process and respects the decay of radioactivity of individual radio-nuclides. These features of the code enables it to evaluate the representative group of decommissioning options as needed and to deliver the numerical data for the decision making process. By scanning the selected input data, it is also possible to find out the reasonable ranges of evaluated parameters. Moreover, the individual decommissioning options or scenarios can be individually optimised using the on-line linked standard software for project optimisation (MS Project). In this way, the code provides the matrices of data which can be used for multi-attribute analysis of decommissioning scenarios.

The basis for these kind of calculations is the facility inventory database which involves the physical and radiological data in an appropriate structure. The paper presents the structure of the inventory databases and the main steps of developing the inventory database. The computer code uses the facility inventory data in direct links to the system for material and radioactivity flow control. The radiological data in the inventory database are automatically updated in relation to the starting dates of individual decommissioning activities, so the calculated data correspond to the real radiological situation of the facility at the time of performing the decommissioning activities.

Dismantling and other technologies for decommissioning and conditions for their implementation can impact cost and other decommissioning parameters like manpower and dose for personnel. Application of various decommissioning techniques is also the subject of optimisation of decommissioning options. The paper presents the methods for selecting the optimal technologies of dismantling and waste management in evaluated decommissioning options. The case of implementation of remote controlled techniques, as the techniques having the largest impact on decommissioning parameters (cost, manpower, dose), is discussed in more detail.

Safety of performing the decommissioning activities should be also demonstrated. This means that the calculated safety related parameters should meet the relevant limits for any decommissioning technique implemented. Critical to decommissioning, from the safety point of view related to personnel, is the dismantling. The methodology is presented to evaluate this, already in the phase of decommissioning planning, the basic parameter being the annual dose limit for an individual member of the staff. Normally, the doses to workers are calculated conservatively. The paper also presents the methodology for evaluating the dose to individuals more realistically.

Waste management issues in decommissioning decision process are normally related to evaluating the representative scenarios of waste management available for the decommissioning project. The process of evaluation in current methodologies is separated from the calculation of parameters of dismantling and other decommissioning techniques. The paper presents the methodology of evaluating the waste management parameters and selection of waste management technologies. The methodology uses the direct data links to the inventory facility database. The paper also presents the methodology for evaluating the impact of conditional release of materials from decommissioning as a special case of the waste management scenario.

The overall goal of the subproject is to demonstrate that the decommissioning computer code with one compact calculation structure including the inherent system for flow control of materials and radioactivity directly linked to the facility inventory database can contribute to better understanding of how to compare and select decommissioning technologies and decommissioning scenarios in an optimal manner. The additional goal, as presented in the paper, is to demonstrate the methodology for harmonising the structure of decommissioning costs as they are presented in the common document of IAEA, OECD/NEA and European Commission [5]. The common use of the cost structure, as presented in this document, will improve understanding of individual cost items involved in decision making process.

2. Scope of the subproject

The scope of the sub-project "Comparative analysis of decommissioning technologies based on model calculations and multi-attribute analysis of specific decommissioning cases of nuclear facilities" as originally proposed for the Coordinated Research Project T2.40.07 and as modified in the frame of the CRP is to present the possibilities introduced into decommissioning costing and planning by development of the decommissioning code OMEGA. The following aspects are presented in the paper:

- The decommissioning code OMEGA as a new tool for general application in decision making process and for planning in decommissioning. The tool implements the internationally accepted standardised structure of items for decommissioning costing

- Evaluation and optimisation of decommissioning options for the purpose of selection of the optimal decommissioning option

- Modelling of dismantling techniques with special attention to application of remote dismantling techniques

- Evaluation of safety in decommissioning

- Waste management scenarios including the analytical approach for evaluation of conditional release of metallic materials

The above listed approaches were supported by model calculations using two model databases.

3. The OMEGA code and its applications

3.1. Basic properties of the code

The methodologies for evaluating cost and other decommissioning parameters were developed based on experience derived from real decommissioning projects and the developed methodologies were then used for similar facilities after adjustment of unit factors and other elements of developed methodologies for the differences in facility size, inventory, local factors and other factors. The quality of results depends on quality of adjustment of unit factors and on the quality of the inventory database. Decommissioning waste management issues in current methodologies are normally evaluated in modules separated from dismantling and other waste generated activities. Another aspect of most of current methodologies is the fact that the cost structure is in general different for various projects and the costs are therefore less comparable, if at all. The methodology for evaluating and optimising the cost and other decommissioning parameters, as implemented in the computer code OMEGA, is based on calculation modelling of the complete decommissioning process including the waste management [6]. This approach improves the limitations of traditional costing methodologies. Main features of the code are following:

- The calculation structure implements in full extent the standardised cost items structure for decommissioning, issued commonly by OECD/NEA, EC and IAEA in 1999 [5]. The calculated costs are transparent, traceable and comparable with other projects.

- Calculation process is sequentially linked-up in such a way that it simulates the real decommissioning process flow in time, and the relevant material/radioactivity flow. The calculation items are linked to the material and radiological data of the inventory database and to the database of interim material/radiological items generated during calculation. In this way, the calculation process uses the actual material and radiological data.

- Calculation process is nuclide-specific and respects the radioactive decay of individual radio-nuclides. This enables the use of the nuclide-specific limits for treatment, conditioning, disposal, release of materials and other specific decommissioning activities within the material flow. This enables the study of the effects of time in the option of deferred decommissioning. The decommissioning infrastructure is simulated by various scenarios for waste management. The scenarios include decommissioning activities linked from dismantling up to the disposal of conditioned radioactive waste or release of materials.

- Calculation structure of the code is standardised for all calculation cases. The extent of individual calculation cases is easy to define. The decommissioning work breakdown structure (WBS) of individual cases is project specific and in the computer code is constructed by grouping and/or linking of the items of the standardised calculation structure to the WBS items. The resulting WBS is generated by the code as the Gantt chart of the decommissioning option. After optimisation in the MS Project software (tasks linking, critical path definition, period dependent activities definition, adjustment of deferred decommissioning phases, etc.), the decommissioning parameters are automatically recalculated according the optimised start dates of individual decommissioning activities.

The developed methodology has a “multiple options” character. This means that several decommissioning options are defined for a decommissioning project in order to evaluate possible scenarios of decommissioning in the frame of the project. Each decommissioning option is calculated, optimised and evaluated individually and the project optimal option can be selected based on multi-attribute analysis. The calculation methodology includes evaluation of exposure of personnel on the level of individual professions involved in decommissioning. Effect of application of remote dismantling versus manual dismantling can be evaluated for optimising the exposure.

An additional aspect of the developed costing methodology, which is related to internal linking of the calculation process and due to compactness of the calculation structure, is the possibility to perform a sensitivity analysis. This can reveal the margins of decommissioning costs and other parameters by considering various levels of contamination, various nuclide composition (effects of alphas), application of various decommissioning technologies, various durations of deferred decommissioning phases, etc.

The code was developed in the period 1999-2003 in the frame of the project for technical support of decommissioning of A1 NPP in Slovakia. The code was tested and upgraded in the period 2004-2005 in a series of model calculations. The principal scheme of the code OMEGA is presented in Figure 1.

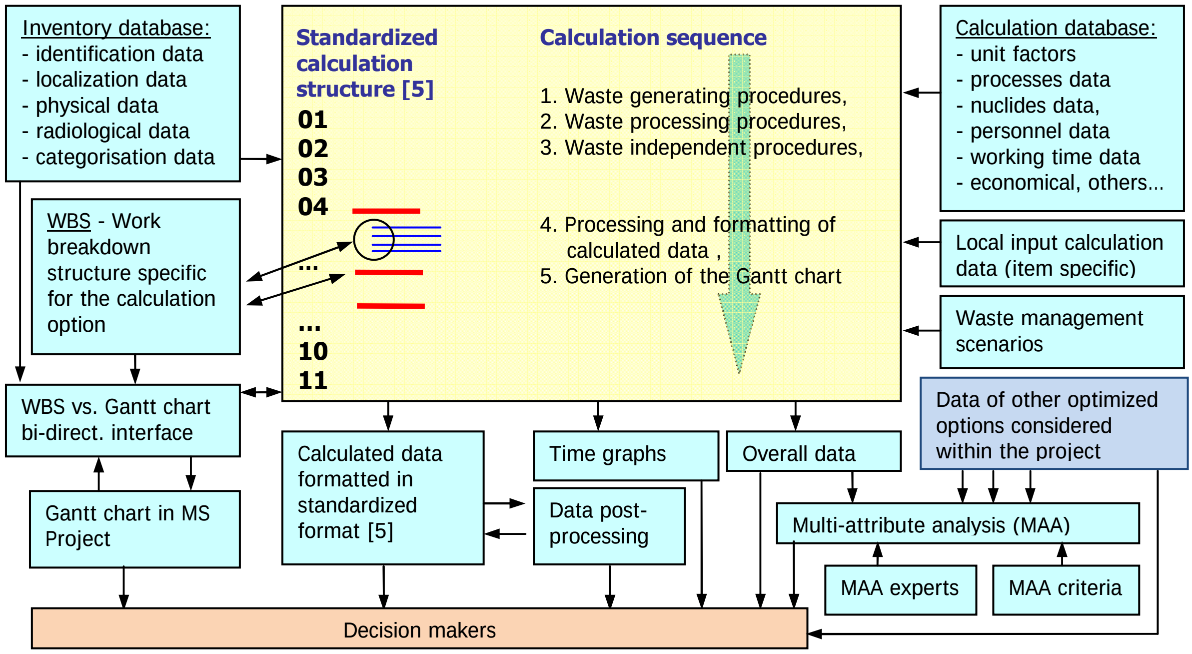

Fig. 1. Principal scheme of the decommissioning calculation code OMEGA.

Until now, the code was applied in evaluating and optimising the decommissioning projects for NPP's in Slovakia (A1 NPP, V1 NPP, V2 NPP, EMO 1,2, EMO 3,4). The code was applied also in evaluating the safety related parameters for normal planned decommissioning activities within the IAEA project DeSa [4], in model calculation for the Swedish Nuclear Inspectorate, for model calculation within this CRP, and for upgrading of the decommissioning plan for Paks NPP in Hungary. Model calculations were realised in the frame of continuous development of the OMEGA code in order to develop new modules for waste management scenarios and for management of uncertainties in decommissioning costing.

3.2. Approaches implemented in the code

The principles of decommissioning costing implemented into the Omega code are following:

- *What to do* (*management of the standardised calculation structure*): configuration of decommissioning activities of a decommissioning option in standardised format using the templates of the standardised structure, generation and management of executive calculation structures, which correspond to the facility structure of buildings-floors-rooms-items in rooms.

- *How to do it* (*management of calculation conditions*): allocation of calculation procedures into the standardised calculation structure, definition of the conditions for calculation, adjusting of the radiological condition at the start of individual activities in order to select manual/remote operations, relevant protection of the personnel and to calculate the real exposure of the personnel; generation/editing of relevant calculation data and correction factors for manpower calculation.

- *In what sequence* (*management of material and radioactivity flow in decommissioning*): implementation of the concept of material and radioactivity flow modelling in decommissioning process based on data linking of the calculation process and on definition of the calculation sequence which corresponds to real primary and secondary waste generation and flow in waste generating decommissioning activities and in waste management activities including the final waste disposal and material release, and effluents from the process into the environment.

- *At what time* (*management of time in decommissioning*): implementation of the concept of on-line optimisation of decommissioning schedules which includes the method for definition of the option-specific work breakdown structure (WBS) and the mutual linking of the WBS items to grouped or non-grouped items of the standardised calculation structure. The time structure as defined in the optimised decommissioning schedule is used for recalculation of costs and other decommissioning parameters in order to evaluate the effect of time on decommissioning.

The first two steps can be identified in each decommissioning costing methodology. The last two steps represent the new items in decommissioning costing methodology developed for the OMEGA code. One of the most important features of the code is the compactness of the standardised calculation structure which includes all activities of the decommissioning option and the activities for management of waste resulting from decommissioning. The full extent of activities of the standardised calculation structure [5] includes also the activities of the transition period after shutdown and the activities of spent fuel management as well.

The character and level of radioactivity can have in general an essential influence on choice of decommissioning activities (manual/remote, etc.), on amount of waste and its processing, on demand of volume in repositories, etc. and finally on decommissioning costs, exposure, manpower and other decommissioning parameters. These aspects are especially important in the case of a facility with non-standard radiological situation, for example after an accident. Material and radiological aspects in the developed standardised decommissioning costing refer to the following issues:

- Algorithmization of the entire material flow in decommissioning by calculation modelling in order to optimise the decommissioning process and waste management

- Algorithmization of radiological aspects in order to identify the location of radioactivity linked to the material flow and to manage the effect of time in decommissioning (to implement the decay of radioactivity at the level of individual radio-nuclides)

The compactness, internal linking of the standardised calculation structure and proper sequencing of the calculation process [6] enables calculation of costs and other decommissioning parameters for a decommissioning option within one calculation run. This includes processing of the calculated data and generation of output data formats and the decommissioning schedule of the option in the form of the Gantt chart in MS Project (Fig. 1). These principles enable an increase in the accuracy of calculation of decommissioning parameters related to material flow in decommissioning, and permit to optimise the waste management and to perform sensitivity analysis.

The concept of nuclide vectors was used in definition of the nuclide composition of contamination, activation, mass/volume activity and dose rate. The nuclide vectors are stored with the date of their definition. Prior to application in calculations, the nuclide composition is recalculated for the decay of individual radio-nuclides. The recalculated nuclide vectors are then used for generation of contents of individual radio-nuclides, used in the nuclide-resolved calculation process. The effect of deferring the decommissioning activities can be analysed using this concept. The concept is applied in calculation of exposure of personnel, in selecting manual or remote operation and in waste management.

The aspect of standardisation of the decommissioning calculation structure has high importance, due to a demand to promote the use of standardised costs structure as agreed by IAEA, OECD/NEA and the European Commission [5]. The templates of the standardised calculation structure were developed for the OMEGA code for automatic generation of the standardised calculation structure and for generation of the default values of input calculation data.

3.3. Calculated data

The computer code OMEGA generates the following groups of calculated parameters:

- General decommissioning parameters: costs, manpower, exposure, duration, number of workers, material and technical media consumption items

- Material parameters: parameters of interim and output materials (weight, inner / outer surface, volume, etc.) including all interim waste forms, final waste form (overpacks), released materials, gaseous and liquid effluents

- Nuclide parameters: radioactivity of interim and output material. Items are evaluated for individual radio-nuclides linked to the material items

- Profession resolved parameters: manpower and exposure is resolved for individual professions and working groups

- Planning parameters: start, duration of defined decommissioning phases, individual decommissioning activities, Gantt chart, equipment needed for performing of individual decommissioning activities

The data formats of calculated data are the detailed data on the level of the calculation item and summary format for the main decommissioning parameters used. Also, for multi-attribute analysis for selection of the optimal option, standardised data formats according the structure in [5], time graphs in pre-selected time scale for any calculated parameter, and the Gantt chart of the decommissioning option in MS Project software is generated.

3.4. Application of the code

The computer code OMEGA [7], [8] is a generic tool for application in decommissioning decision making processes, planning of decommissioning of nuclear facilities of various types, radiological properties and systems and structures and for optimisation of waste management. The following applications are available:

- Definition of the set of decommissioning options for the facility to be decommissioned, covering all possible scenarios to be evaluated. The scenarios can include immediate or deferred options, various waste management scenarios, remote vs. manual dismantling etc.

- Standardised calculation of costs and other decommissioning parameters for individual calculation options, processing and evaluation of calculated data

- Optimisation of individual calculation options within the individual options – optimisation of the Gantt chart, and safety related parameters and optimisation of waste management

- Comparison of decommissioning options and selection of the optimal decommissioning for the given decommissioning project based on multi-attribute analysis

Another aspect of multi-option work is the management of parallel projects of decommissioning which can be co-optimised by generating the joint Gantt chart. This enables one to evaluate the mutual interactions of on-site parallel projects and optimise the use of common facilities and resources.

3.5. Input data for applications using the OMEGA code

There are three basic groups of data used as the input data for applications with the OMEGA code (see also Fig.1):

- Facility inventory databases - hierarchical database system of the nuclear facilities, buildings, floors, rooms and equipment in the rooms

- Database of calculation parameters - unit factors and other parameters of processes, working groups and professions, working time structure, radiation protection, radio-nuclides, general technical-economical parameters

- Input data specific to individual calculation items — general input data defined for each item of the calculation structure (e.g. increase factors, fixed costs items, number of working groups, shift work, etc.), calculation item specific data (e.g. duration, working group and working conditions for individual period-dependent activities)

The general structure of the inventory database is the hierarchical structure starting on the top at the list of nuclear facilities and going down via linked structure of floors, rooms and equipment identified in individual rooms. The equipments are the elements of systems, elements of building surfaces and elements of structures (building materials). The data defined for each items at this level are the identification data (name, allocation to systems, …), localisation data (allocation to rooms), physical data (mass, external and internal surfaces, inner volume) and radiological data (level of external and internal contamination, mass radioactivity, dates of definition of data, nuclide vectors). Nuclide vectors used for definition of radiological parameters represent the individual radio-nuclide content, normalised to a value of 1.

Two model facility inventory databases, used for model calculation in the frame of this CRP sub-project were developed based on real data of two types of NPP's: a PWR reactor and a gas cooled and heavy water moderated reactor. The first, the NPP, has excellent parameters in operation. In the second case, an accident occurred during operation.

The database of calculation parameters and the data specific for individual calculation items used for model calculations in the frame of the CRP project, were the data used generally in the OMEGA code. These data are localised for conditions in Slovakia including the data for waste management.

4. Procedures for evaluation and optimisation of decommissioning options

4.1. General procedure for evaluation of decommissioning parameters

The main decommissioning parameter is the cost. The methodologies for evaluation of decommissioning parameters are normally considered as costing methodologies, because the parameters other then cost are ultimately used for calculation of cost. The standard procedure for calculation of cost and other decommissioning parameters, as identified in major decommissioning projects, is based on the unit factors approach having the following main steps [9]:

(1) *Definition of cost categories:* defined as the activity-dependent costs, related to the extent of "hands-on" work like dismantling, the period-dependent costs, proportional to duration of individual activities/phases and the collateral costs and costs for special items which can neither be assigned to hands-on work activity nor to period-dependent activity.

(2) *Identification of decommissioning activities and inventories:* identification of discrete elementary activities for which unit factors are defined and completion of the list of activities within a facility buildings/equipment inventory in order to define the overall extent of activities.

(3) *Definition of unit factors:* unit factors are defined in accordance to the details of the items considered in the plant inventory and in the decommissioning activities listing. Unit factors are defined for ideal working conditions and correction factors are defined that reflect the specific working conditions (radiation, working height, etc.).

(4) *Project scheduling and staff requirements:* project time schedule is constructed based on calculated duration of individual hands-on work phases and based on the plant inventory data. This is the base for identification of the critical path for decommissioning activities. Calculated duration of decommissioning activities / phases is used as a basis for definition of duration of period-dependent activities for which the staff is defined.

(5) *Definition of collateral costs and costs for special items*: Definition of fixed costs like cost for heavy equipment for site support, health physics equipment and supplies, licenses and permits, costs for lighting, heating, cooling, income from sold equipment or scrap, etc.

(6) *Total costs definition:* Total cost estimates are obtained as a sum of the costs for three categories: activity-dependent costs, period-dependent costs, collateral costs. The cost estimates may be adjusted to include a contingency that reflects the level of uncertainty in the estimates. A separate contingency expressed in some special cost items may be applied to the total cost estimate for the processes with high uncertainty.

This principal procedure is implemented also in the computer code OMEGA, enhanced by the third and fourth steps presented in 3.2.

4.2. Generation of the calculation structure

The first step is the definition of the calculation option, depending on the type of scenario selected (involving decommissioning activities in proper extent) and on the facility (type and inventory). As for the decommissioning activities, the approach implemented in the OMEGA code is to use the standardised structure as defined in [5] and extend to more detailed levels, as the base for the pool of decommissioning activities. This detailed standardised template is used for all calculation cases in the OMEGA code. The extent of real calculation is defined in each calculation option by clicking-in the relevant calculation items. In this way the flexibility of a universal calculation base was achieved.

The pre-requisite for effective work with the OMEGA code is the inventory database of the facility with relevant systems, buildings and radiological data, and the calculation database with relevant data for processes, profession / work time data, material / nuclide data and other data (See 3.5). The process of generation of the standardised calculation structure has three steps using the templates which facilitate the work of the user. The base for this work is the previously-presented standard template which covers the decommissioning activities as defined in [5]. In the first step, based on the general standard template, the user can develop the master template which is specific for the type of a nuclear facility. In the second step the user can modify the selected master template to the standardised structure specific to the decommissioning option to be calculated. In this step the user can define as many calculation options as required for the evaluation within the decommissioning project. The option-specific standard structure of decommissioning activities involves also the prescription for generation of lower levels of calculation items i.e. for allocating in accordance with calculation procedures and the calculation sequence.

The third step is the automatic generation of the executive standardised calculation structure used for calculation. The generation involves generation of a calculation-row corresponding to the inventory database. The typical feature of this structure is that it has the hierarchical structure of the buildings – floors – rooms/cells – inventory items in the room/cell in selected sections of the standardised structure. The generated structure contains also input calculation data with default values. After the generation, the user can review/edit the generated calculated structure and the generated default values of the calculation data and can define the extent of calculation by clicking in the individual calculation items. An example of the executive calculation is in Figure 2. Automatic generation of the executive calculation structure is a strong feature of the OMEGA code. It enables one to generate as many calculation cases as are needed to be evaluated before a decision on the optimal option. The generation is fast and takes only a few hours for one calculation case.

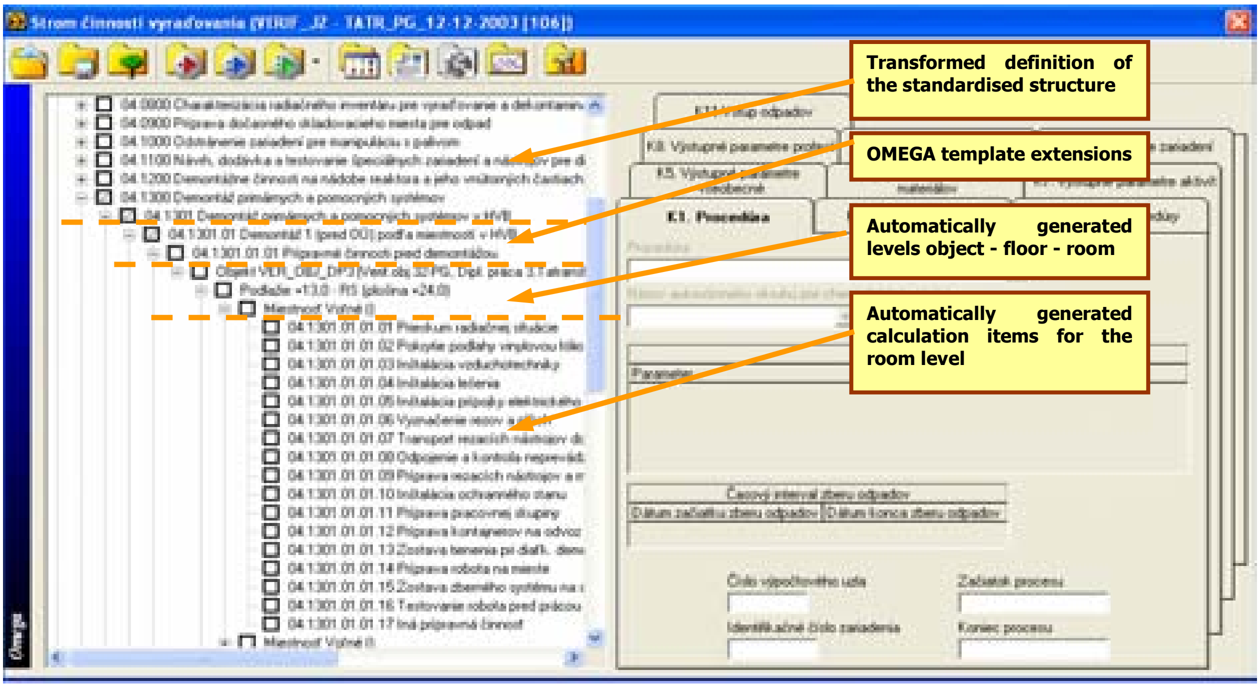

Fig. 2. Example of generated standardised calculation structure, with delineating the levels of details of calculation items.

4.3. Definition of calculation conditions for the options

The second step involves making the calculation structure ready for calculation. This means the structure should have allocated proper calculation procedures, default input data and links to the inventory database. This is done during the generation of the executive calculation structure automatically. Calculation procedures for the individual calculation items or formulas for generation of the calculation procedures are pre-defined in the standardised calculation templates. The executive calculation procedures are allocated automatically during the generation of the executive calculation structure as one default procedure and as a set of supplemental (possible) calculation procedures for editing by the user.

The remote controlled / manual procedures are selected automatically based on actual radiological inventory data and acceptance limits for manual work. The correction factors (work in radiation fields, on scaffolding) are calculated during generation of the executive calculation structure based on inventory data and can be edited by the user after generation. The local input data entry points (e.g. working group parameters / duration of period dependent activities, fixed costs, contingency, etc.) are generated as default values or empty data entry fields during generation of the executive calculation PSL structure. The user enters or edits the input calculation data. User friendly tools for calculation data editing / entry were developed.

4.4. Management of material and radioactivity flow

The calculation sequence during the calculation of the data has high importance due to the internal system for management of the material and radioactivity flow. It is necessary to set the proper sequence. Because the system uses the material data in the on-line manner, the material data generated in waste generating procedures (like dismantling) are used as the input data for waste processing procedures (like sorting).

The concept of material and radioactivity flow control in decommissioning developed at DECOM, a.s., represents an original generic methodology. Tools are implemented in the standardised OMEGA code for on-line optimisation of decommissioning and waste management processes. The modelling of the processes is based on mathematical partitioning of material inventory into one-material elements which enter into the pre-defined sequence of calculation and sorting procedures, linked to each other by unambiguous material links. To each one-material calculation element are linked radiological parameters which are generated during the material partitioning. The generation is based on the calculation category of the inventory items and distribution coefficients relevant for the item category. The generation of secondary waste is considered.

The concept of nuclide vectors is used for definition of radiological parameters of inventory items (normalised participation of individual radio-nuclides in radiological inventory data). The linked radiological parameters are recovered dynamically during the calculation to the start dates of individual decommissioning activities. The decay of radioactivity of individual radio-nuclides is respected through the entire decommissioning process. The calculation procedures implement the parameters of individual processes of the decommissioning infrastructure (actually available or planned) for the decommissioning project.

The sorting procedures implement the limits for releasing of materials, acceptance limits for disposal of materials, acceptance limits for individual process (if they are defined) and parameters of individual processes which affect the material / radiological parameters of evaluated items. Both types of procedures can be linked to pre-defined scenarios of waste management for decommissioning starting from pre-dismantling decontamination up the release of materials or disposal of materials in the surface repository or deep geological repository.

Multi-stream material calculation structures can be defined by appropriate sequence definition combined with selected sections of Proposed Standardized List (PSL) calculation structure and by definition of dates for waste entry into individual material streams. Keyboard data entry of waste items for general application in waste management modelling and optimisation is feasible. The waste streams are optimised in the standard Microsoft Project software.

The principle of material and radioactivity flow control is presented in Figure 3. This principle is implemented into the OMEGA code in order to manage the material flow in decommissioning within one compact calculation structure.

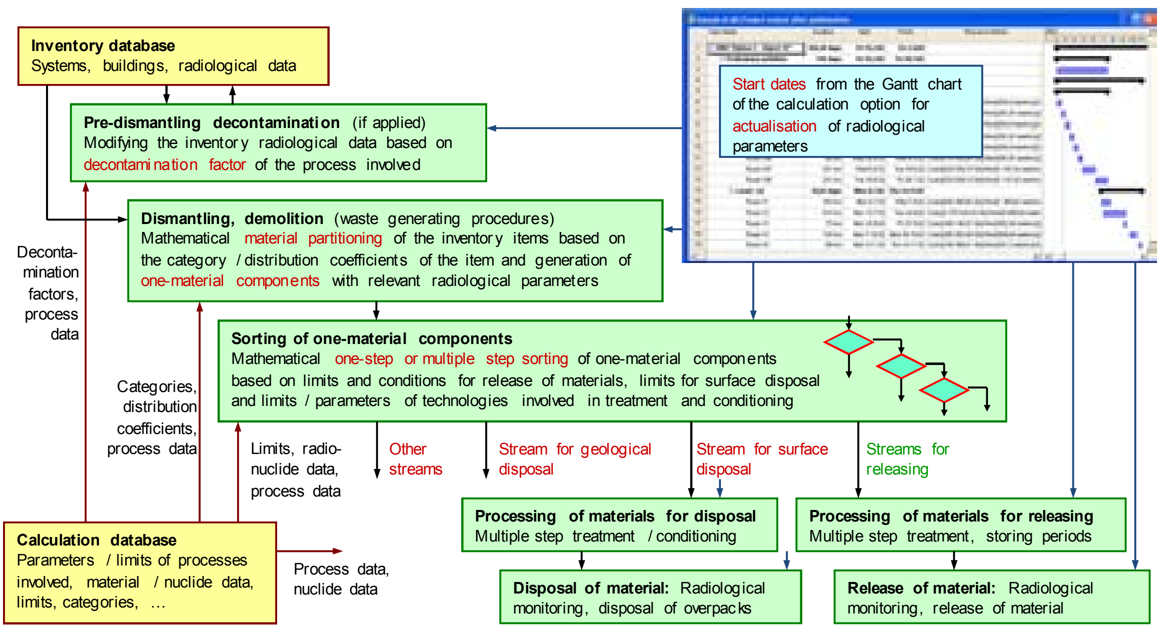

Fig.3. Principle of the material and radioactivity flow control as implemented in the OMEGA code.

4.5. Optimisation of decommissioning options

After calculation of the data, it is necessary to develop the decommissioning schedule for the calculated option. The concept of generating the decommissioning time schedule in the form of the Gant chart in Microsoft Project with bi-directional data link to the calculation structure was implemented in the OMEGA code. This enables the on-line optimisation of the decommissioning time schedule, and this concept enables one to implement directly the impact of time (decay) on decommissioning parameters dependent on radiological parameters like contamination, activation and dose rate.

The real sequence and structure of activities in the work breakdown structure (WBS) of a decommissioning project is different from the standardised structure. The WBS in OMEGA can be defined individually for each calculation option as a structure of project items. The WBS can be linked to the standardised calculation structure by specific grouping or linking of items of the standardised calculation structure to the items of the WBS. The developed interface enables one to transform the WBS into the Gantt chart in the Microsoft Project software including the project data.

The Gantt chart can be optimised using the standard tools of Microsoft Project. The optimisation represents the linking of activities with calculated durations (activity dependent types of tasks, like dismantling) in order to define the critical path for the option, adjustment of duration of period dependent activities and modifying the optimisation parameters (number of working groups or shifts) where needed. The optimised Gantt chart is used for transfer of start dates and durations back to the executive calculation structure. The start dates for the grouped items are sequenced for individual de-grouped items. Linked items are transferred directly.

The start dates of individual calculation items, derived from the Gant chart of the calculation option, determine the actual values of radiological parameters involved in the calculation item. There are two basic modes of calculation run in OMEGA. In the first calculation run, all calculation items start with the same date. This mode is used for first generating the Gantt chart before its optimisation in Microsoft Project. In the second and nth calculation run, the start dates are derived from the optimised Gantt chart. Effects of time decay (e.g. in deferred dismantling) can then be evaluated directly. The overall cycle of optimisation of decommissioning option using the Gantt chart is presented in Figure4.

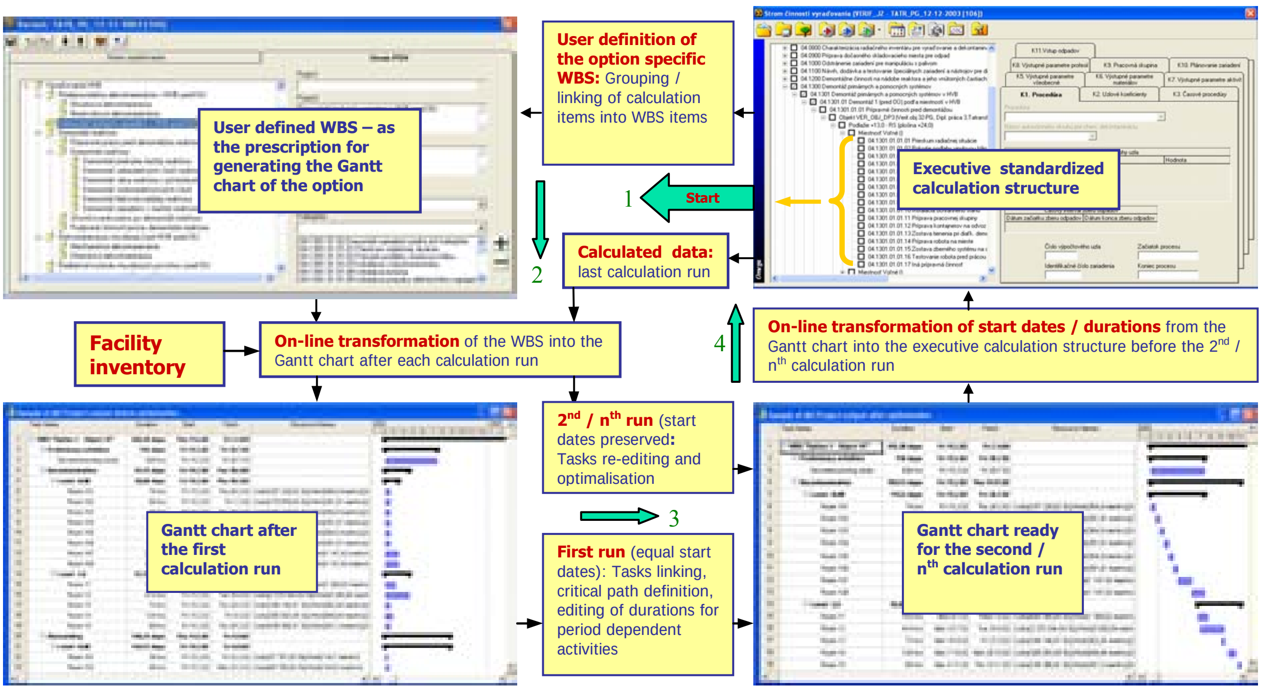

Fig.4. The procedure for optimisation of decommissioning options using the Gantt chart.

4.6. Summary of style of work with the OMEGA code

The work with OMEGA has an iterative character with following main steps (Fig.5):

- Calculation of parameters in the first calculation run with equal start dates
- Generating formats of calculated data and Gantt chart in MS Project
- Optimisation of Gantt chart in MS Project (linking, critical path, etc.,)
- Load of start dates / durations from optimised Gantt chart into OMEGA, modification of optimisation parameters in the calculation structure
- Calculation of decommissioning parameters with start dates derived from the Gantt chart

This cycle can be repeated with start dates derived from the Gantt chart up to achieving the final optimised decommissioning option ready for multi-attribute analysis. The same procedure is realised for all calculation cases of the decommissioning project and the multi-attribute analysis is done using the files of overall data for each calculation case.

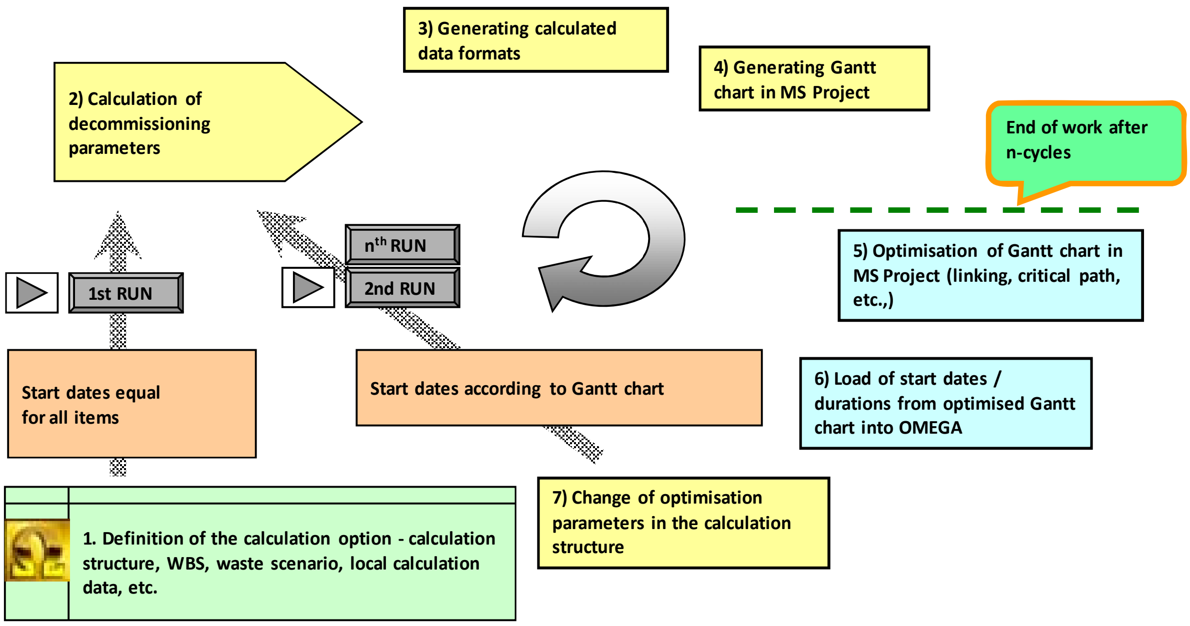

Fig.5. Graphical interpretation of main steps of the iterative work with the OMEGA code.

5. Optimisation of dismantling techniques

One of the goals of the CRP is to identify the methods for optimising the techniques for decommissioning. Critical techniques from the safety point of view and considering the aspects of complexity are the dismantling techniques. This chapter presents the approach to algorithmisation of dismantling techniques in the OMEGA code and methods for optimisation of application of various techniques including application of remote techniques, in the phase of planning the decommissioning.

5.1. Approaches to modelling of dismantling activities

The principle of algorithmisation of dismantling techniques in decommissioning used in the OMEGA code is the modelling of the real sequence for performing the individual elementary dismantling activities. The real sequence in dismantling is composed of three basic groups of activities:

- Activities for preparation of dismantling at the working place of dismantling (individual rooms)

- Dismantling of individual items according the content of the database related to the working place
- Activities for finishing the work at the given working place in order to prepare the working place for next decommissioning activities
- Supporting activities in the case of complex systems for dismantling like reactor dismantling

The preparatory activities represent the set of activities realised in order to establish proper and safe working conditions at the dismantling working place. Calculation modelling of this phase is the allocation of a set of elementary activities. The set of these allocated activities is then adjusted by the user in order to accommodate the extent of preparatory activities to the conditions at the given working place. Activities of this group are normally activities with fixed manpower per activity or where the manpower is dependent on the size of the working place (individual rooms).

The dismantling phase represents the set of dismantling activities according to the content of elementary inventory items at the working place. The inventory items are individual elements like tubes, valves, motors and similar components. Activities of this group are typical inventory-dependent activities. The inventory items are classified according the typical decommissioning category. The principle of categorisation defines to typical representative types of equipments (for example pipes made of stainless steel with defined diameters) for which the unit factors are defined. Typical unit factors are manpower (man-hours needed for dismantling of a normalised mass of equipment of given dismantling category, consumption of electricity per normalised mass, etc.). This principle reduces significantly the extent of unit factors needed for calculation of decommissioning parameters and makes the calculation effective.

The finishing activities represent the set of activities to be performed after the dismantling of the last item in the inventory database. The purpose of these activities is to leave the working place in a clean and safe condition ready for application of the next activities like decontamination of building surfaces. The calculation procedure is the allocation of activities, normally the inverse of preparatory activities. The procedures of adjustment by the user and the types of activities are similar to preparatory activities.

The supporting activities are specific for the application of complex dismantling systems. An example of this type of activities is the maintenance of dismantling systems during dismantling or continuous radiation monitoring at the working place.

The above presented principle of algorithmisation of dismantling activities represents the implementation of the “bottom-up” principle which is considered as the most accurate method for evaluation of decommissioning parameters [2]. The data are calculated at the lowest level of details of decommissioning activities and the results are consequently grouped up to the level of overall results.

The optimisation of dismantling activities includes the generation and application of increase factors for activities performed under non-ideal conditions. The unit factors for dismantling are defined for ideal working conditions, i.e. where no restrictions are present. In non-ideal conditions, like dismantling in areas with higher dose rates using personal protection means, working on scaffolding, in congested areas (the most frequent examples), the increase factors are applied in order to increase the manpower needed for performing the given dismantling activity. The increase factors are generated automatically during the generation of the calculation structure based on data in the inventory database and can be modified by the user before the calculation run. See 4.3.

5.2. Selection of dismantling techniques for calculation

The dismantling techniques, whose parameters are used for calculation of decommissioning parameters, are allocated to calculation items during the generation of the executive standardised calculation structure. The main dismantling techniques and their combinations are the following:

- Oxygen - acetylene cutting of general equipment, preferably for those made of carbon steel
- Plasma cutting for general equipments, preferably for stainless steel equipment
- Mechanical cutting by mechanical saw or other mechanical cutting method which does not generate much heat, used in applications where low generation of aerosols is required, or for cutting of equipment with large wall thickness like reactor vessels
- Hydraulic shear-cutting for pipes with small dimensions, electrical cables, components of ventilation ducts and other equipment with thin walls
- Manual dismantling using standard mechanical hand tools
- Grinding for cutting of equipments with medium wall thickness. The technique has relatively high cutting rate, but the release factors for radio-nuclides is high.

Allocation of techniques to calculation items, as applied in the computer code OMEGA, is organised based on dismantling categories defined for inventory items to be dismantled. The allocation is organised according the Table 1. In the table are vertically listed the decommissioning categories and horizontally the applicable techniques. The table shows the default techniques (green colour) as selected by the code during generation of the calculation structure and alternative techniques, generated as other possible techniques which can be selected alternatively by the user in the generated calculation structure.

Table 1. Allocation of techniques to dismantling categories

Dismantling category	HDCT	COBO	PLSM	OCHC	MSW	OACT	PLHC	MNOC	MAND	MAPL	GROC	GRPL
Piping (SS), diameter =< D25 mm												
Piping (SS), diameter over 25 mm												
Piping (CS), diameter =< D25 mm												
Piping (CS), diameter over 25 mm												
Tanks (SS)												
Tanks and containers (CS)												
Heat exchangers (SS),												
Heat exchangers (CS) ,												
Pumps (SS, CS), mass <= 50 kg												
Pumps (SS), mass over 50 kg												
Pumps (CS), mass > 50 kg,												
Ventilators (SS, CS), mass <= 50 kg												
Ventilators (SS), mass > 50 kg,												
Ventilators (CS), mass > 50 kg,												
Valves (SS)												
Valves (CS)												
Electric motors, mass <= 50 kg												
Electric motors, mass > 50 kg												
Air conditioning components - piping (SS)												
Air conditioning systems others (SS)												
Air conditioning components - piping (CS),												
Air conditioning systems others (CS),												
Air conditioning systems, (Al)												
Electrical cables & conductors												
General electric equipment, (CS) mass <= 50 kg												
General electric equipment, (CS) mass > 50 kg												
Thermal insulations, non-metal covering												
Steel constructions, (CS)												
Small piece components, shielding (CS)												
Hoisting equipment (CS), electrical tackles												
Digestors, sampling boxes (CS)												
Piping feedthroughs, gulleys												
Hermetic and shielding doors (CS)												
Stainless steel linings, (SS)												
Carbon steel linings, (CS)												
Other general equipment												
Casing of technological equipment (CS),												
Casing of technological equipment (SS),												

Some techniques presented in the Table are used as combined techniques due to material composition and physical properties of the dismantled equipment. The abbreviations for techniques presented in Table 1 follow:

HDCT	Hydraulic shears cutting
COBO	Core boring
PLSM	Plasma cutting
OCHC	Oxygen cutting - hydraulic cutting (combined technique)
MSAW	Mechanical cutting by saw
OACT	Oxygen cutting (oxygen - acetylene cutting)
PLHC	Plasma cutting - hydraulic cutting (combined technique)
MNOC	Manual dismantling - oxygen cutting (combined technique)
MAND	Manual dismantling (by tools)
MAPL	Manual dismantling - plasma cutting (combined technique)
GROC	Grinding - oxygen cutting (combined technique)
GRPL	Grinding - plasma cutting (combined technique)

The release factors for generation of aerosols during dismantling for dismantling techniques are identified in the Table 1 by colour. These factors are used for calculation of radioactive aerosols released during cutting into the working area. The calculated data are used for evaluation of safety parameters for personnel and for calculation of the gaseous effluents. The values are defined conservatively and represent the fraction of total contamination of the equipment released. They are the following:

PLSM	Release factor = 10 %
OCHC	Release factor = 1 %
HDCT	Release factor = 0,1 %

5.3. Room oriented approaches to modelling of dismantling activities

Depending on the type of the working place, the breakdown of the dismantling activities can be organised basically as the room oriented structure of dismantling activities or the system oriented structure of dismantling activities.

The room oriented structure of dismantling activities is the basic approach used for dismantling of equipment which is located in rooms of small and medium size (not the large rooms like the reactor hall). The equipment within these rooms is mostly the standard decommissioning categories like pipes, valves, etc. for which the standard unit factors can be used (not the complex equipment). The principle is presented in Figure 6.

The set of preparatory, dismantling and finishing is generated for each room registered in the inventory database. The activities are generated in full extent for each room including the calculation procedures and the default input data. The user then selects activities taking into account specific properties or conditions for individual rooms.

Set of preparatory activities:

- covering of the floor with protective foils to inhibit the contamination of the floor
- installation of local ventilation to suppress the aerosols from dismantling
- installation of scaffolding for dismantling activities
- installation of temporary electrical connections and media for dismantling
- delineation of cuts on equipment
- transport of dismantling tools to the room
- isolation of equipment from electrical connection or operating media
- preparation of dismantling tools for the work
- installation of protective tenting for suppressing the spreading of aerosols
- preparation (instructions) of the working group for the work
- transport of containers with dismantled materials
- specific preparatory activities defined by the user

Set of dismantling activities according the ALARA principle:

- dismantling the inventory item with the highest dose rate at the working distance
- sequential dismantling of inventory items according the decreasing dose rate
- dismantling of the last inventory item with the lowest dose rate

Set of finishing activities:

- removal of protective foils on floors
- removal of local ventilation
- removal of scaffolding
- removal of temporary electrical connections and media for dismantling
- removal of protective tenting
- removal of dismantling tools
- transport of containers
- final cleaning of the room after dismantling
- specific finishing activities defined by the user

Fig.6. Principle of modelling of the room oriented dismantling scenario at the room level.

5.4. System oriented approaches to modelling of dismantling activities

A system oriented approach for organising the decommissioning activities is applied mostly for equipment with large dimensions and complex structure, like reactors, refuelling machines, large components of the primary circuit, etc. The procedures are specific for each component and normally the dismantling is the procedure inverse to construction. The structure of decommissioning activities is specific to dismantled system and typically is organised according to the individual construction sub-assemblies of the dismantled system. This procedure is facilitated by the fact that the technical documentation for complex systems like construction, materials, recommended procedures for maintenance, etc., is organised according to the subassemblies.

The approach to modelling of the system oriented dismantling is presented in Figure 7. The set of preparatory, dismantling and finishing activities is repeated in the calculation structure for each construction sub-assembly. The set of preparatory and finishing activities is the same as for the room oriented approach and again the user selects the relevant activities for calculation. Additional specific activities can be defined by the user. In comparison with the room oriented approach, additional sets of activities are defined at the beginning and the end of the dismantling sequence for general preparatory and finishing activities and a set of auxiliary activities parallel to the dismantling sequence, such as continuous supporting activities like radiological monitoring, waste removal, maintenance of dismantling equipment, etc.

Selection of these additional preparatory, finishing and supporting activities depends on the constructional complexity of the system to be dismantled and on the dismantling system used. These specific activities are defined case by case and normally are defined as period-dependent activities for which the duration, the personnel and radiological conditions are defined.

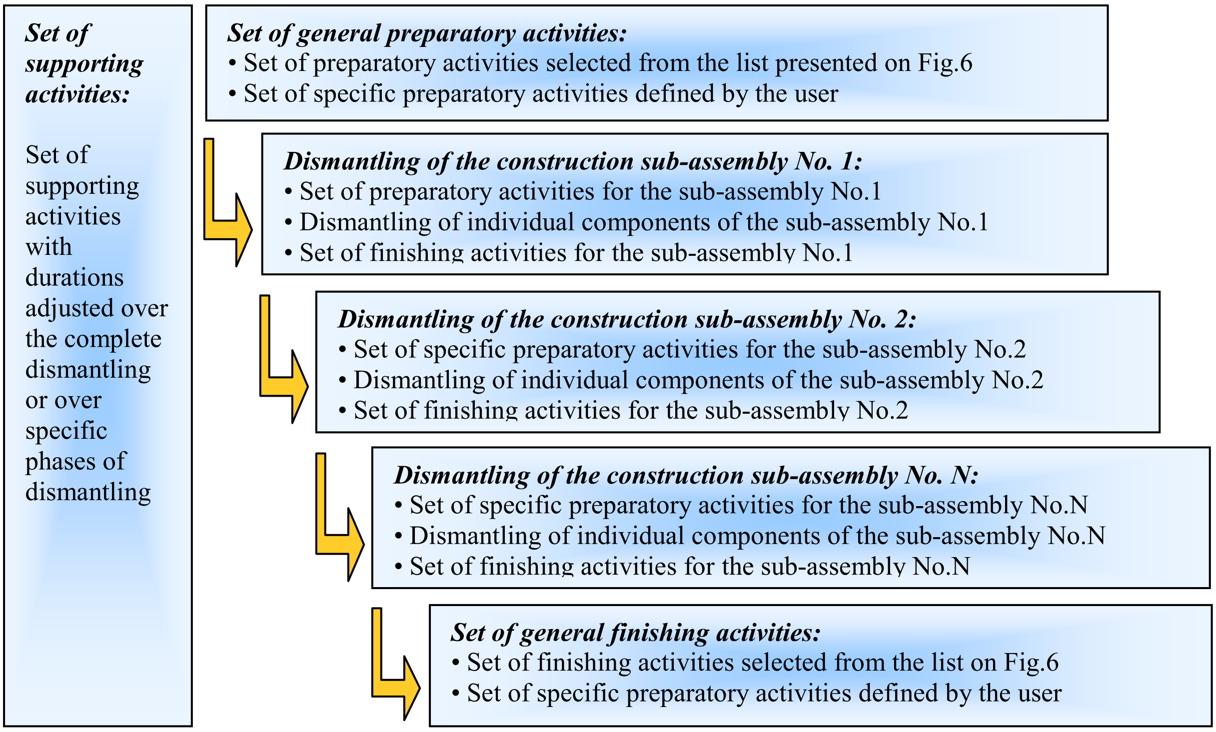

Fig.7. Principle of modelling of the room oriented dismantling scenario.

5.5. Implementation of remote controlled techniques

The remote controlled dismantling techniques are implemented in the case of high dose rates in the vicinity of the equipment. This approach should also take into consideration and used to evaluate the decision making and planning phases of decommissioning. The basic difference compared to manual dismantling are the higher unit factors (by a factor of approx. 5 – 10) and lower exposure of personnel which normally is lower than the annual dose limit for individuals.

The OMEGA code implements the remote controlled techniques automatically by evaluating the dose rate at the start of dismantling. If the dose rate is below the defined limit for implementation of the remote controlled technique, the code calculates the dismantling using the manual techniques. In the case the dose rate is over the limit, the code calculates the dismantling using the remote controlled techniques. The simplified scheme of this procedure is presented in Figure 8.

The extent of preparatory activities is the same as in previous chapters, extended for activities like testing the cutting system, preparation and setting of the system for cutting, preparation of the system for collection of waste and installation of shielding. Similarly, for finishing activities there are additional activities inverse to preparatory activities like dismantling of the shielding, removal of the cutting system and removal of the waste collection system.

The scheme in Figure 8 involves also the case of dismantling outside of the controlled area. The procedure is the same as for the case of dismantling within the controlled area, but the set of preparatory and finishing activities is reduced and does not involve the activities related to radioactivity.

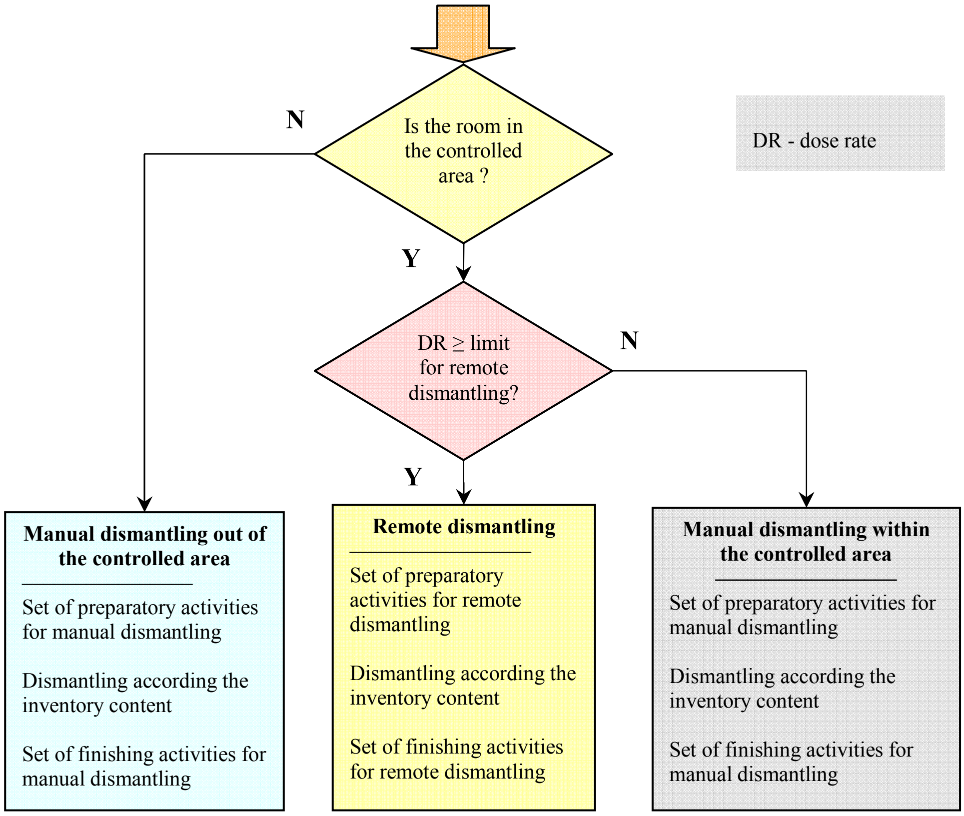

Fig.8. Simplified scheme for implementation of remote controlled techniques.

5.6. Modelling of other dismantling and decontamination techniques

For less extensive sets of preparatory and finishing activities a similar approach can be defined. Also, other typical room oriented dismantling activities like dismantling of auxiliary equipment, dismantling of embedded elements, dismantling of asbestos, dismantling of contaminated or activated concrete as they are defined in the chapter 4 of the standardised structure of items for decommissioning [5] can be addressed using this approach.

Other groups of activities, organised as room oriented activities, include decontamination of building surfaces and radiation survey of building surfaces as the last activities implemented in the frame of dismantling and decontamination. The modelling of these activities is in principle the same as the dismantling. The specific sets of preparatory and finishing activities are defined for each group of activities with a structure similar to dismantling.

6. Examples of cost and cost-benefit analysis using the OMEGA tools

Sensitivity analysis in evaluation of cost and other decommissioning parameters

An example of analysis of cost, manpower and number of final waste packages for LLW/ILW repository as well as waste for a geological repository is presented in Figure 9 [8]. The evaluated activities are dismantling and management of waste from dismantling of selected systems from the reactor of a model NPP with a PWR reactor based on real inventory data. Options 1 and 2 demonstrate the impact of level of contamination, options 3 and 4 the effect of deferring dismantlement, the options 1 and 3 the impact of nuclide composition. Cs 137 and alphas are present in options 1 and 2.

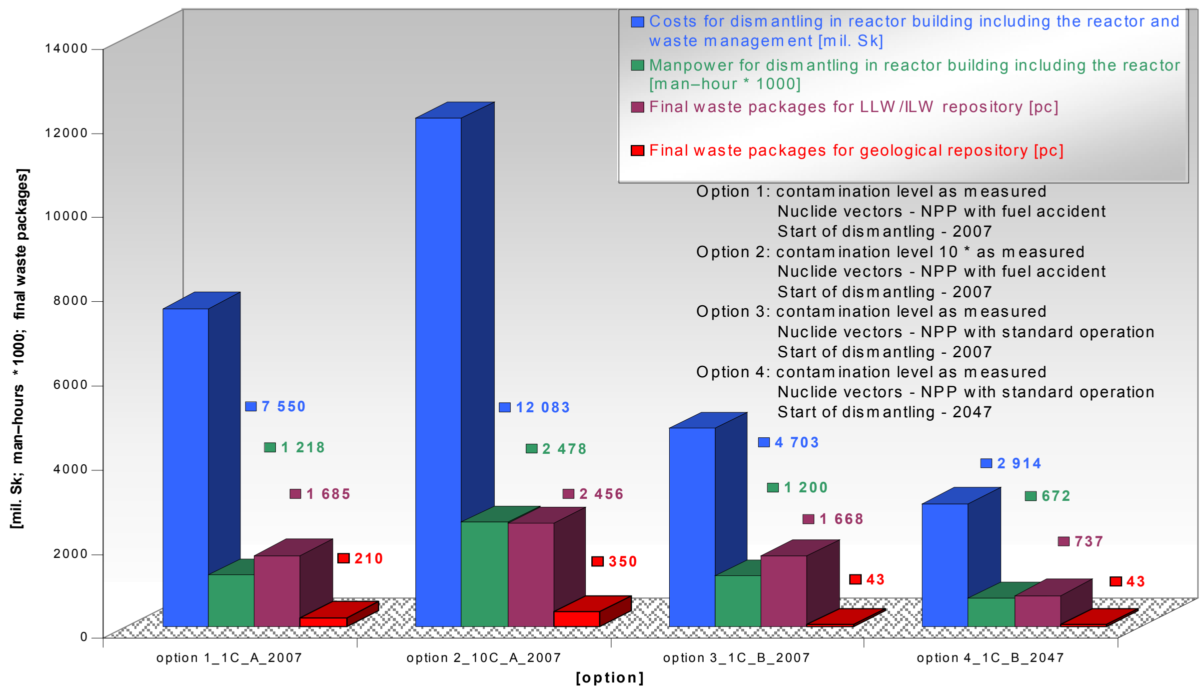

Fig.9. Example of parametric cost analysis using the OMEGA code.

Evaluation of cost (manpower) versus exposure of personnel

One of the methods for reducing the exposure of personnel during dismantling is the application of remote dismantling. In this case, the OMEGA code selects the remote dismantling methods automatically, based on the actual dose rate in the vicinity of the equipment to be dismantled and based on the pre-selected value of the dose rate for application of the remote dismantling techniques. Implementation of the remote dismantling decreases the exposure of personnel due to fact that the personnel is located in shielded working places, but the manpower needed for performing the decommissioning activities is significantly higher (approx. 5-10 times) and the costs for the work are also higher, in proportion to the manpower.

Optimisation of the level of the dose rate at the equipment for implementing remote dismantling can be performed in the computer code OMEGA when all inventory data for the whole NPP are available. The optimization is NPP specific, depending on the real radiological state of the NPP. The methodology of calculation, as applied in the OMEGA code, enables the code to select automatically the application of manual or remote dismantling technique based on:

- Actual dose rate at the equipment to be dismantled (adjusted for the date of start of dismantling)
- Dose rate limit for application of the remote dismantling – defined by the user

The method can be used for evaluation of the optimal level of application for remote dismantling and for cost benefit analysis of typical costs and manpower versus dose uptake during dismantling. Model calculations were performed for the primary circuit of the A1 NPP in Slovakia. The results are presented in Figure 10. The results show that the optimal level for application of remote dismantling is in the range of 100 - 200 micro Gy/hour. The individual dose uptake for each member of the working group can be optimized also by varying the number of working groups in order to meet the annual limit of 20 mSv, depending on duration of the process of dismantling.

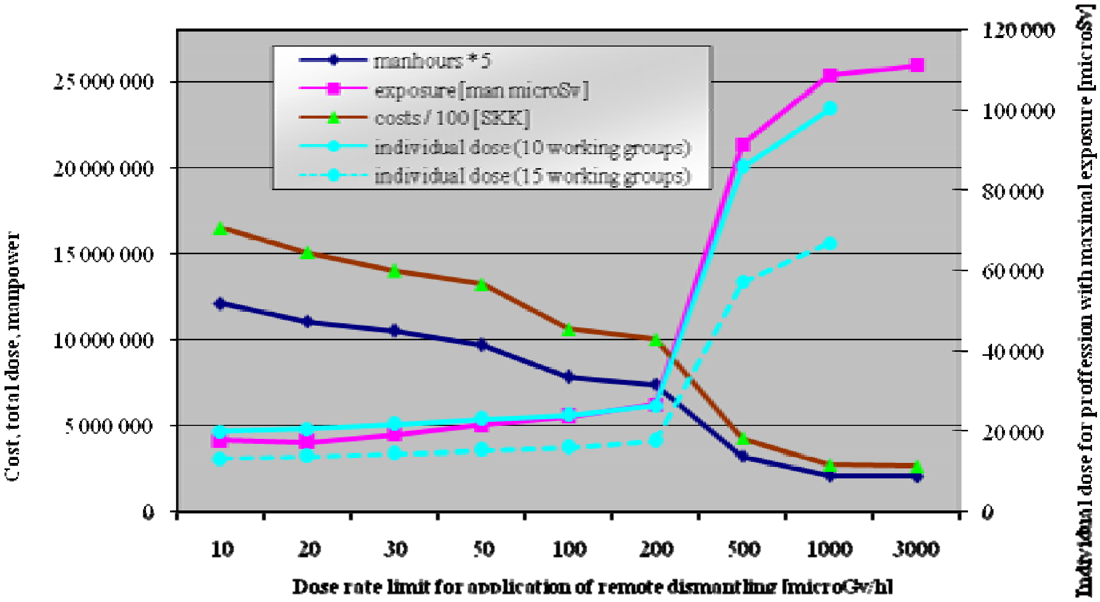

Fig.10. Evaluation of costs for dismantling of selected sections of the primary circuit of A1 NPP (Slovak Republic) versus dose uptake by varying the limit for application of remote dismantling.

7. Evaluation of safety in decommissioning

7.1. Safety related parameters in decommissioning calculated by the OMEGA code

From the safety point of view, the following parameters are needed for evaluation of the safety of decommissioning activities:

- Dose to personnel
- Gaseous effluents and dose to public
- Liquid discharges (waste water from the site) and dose to public

Dose to personnel takes into consideration a calculated duration of individual planned decommissioning activities and radiological conditions at the working place and both external and internal exposure. Individual radiation protection means are taken into account. Optimisation of the dose uptake can be accomplished by controlling the number of personnel for performing the work, by performance of pre-dismantling decontamination of systems or by implementation of remote techniques. The calculated data in the decision making and planning phase are used for demonstration that the planned decommissioning activities can be realised within the limits of annual exposure of individuals.

The data for gaseous effluents are calculated in order to demonstrate that the impact of planned decommissioning activities on the critical group of public is within the limits of exposure, based on existing exposure pathways for evaluation of migration of radio-nuclides to the critical group under local conditions. The gaseous radioactivity effluents are calculated for individual radio-nuclides at the discharging point of the central ventilation stack of the nuclear facility. These data are compared with the authorised limits of gaseous effluents for the site. Gaseous effluents are included in the waste management system of the OMEGA code and are calculated as one of the characteristic items of the material and radioactivity flow control systems.

The data of liquid discharges are calculated as volumes of discharged waste waters from the decommissioning activities. The discharging of waste waters is limited according to individual radionuclides. The values of limits are derived from facility specific scenarios for the critical group for radio-nuclide intake under local conditions. If it is demonstrated that the specific radioactivity of discharged water is under the limiting value, then, the dose uptake of the public is under the limiting value. The setting of the system for material and radioactivity flow in the OMEGA code is defined in that way: that only water under the authorised discharging value is discharged.

Based on these considerations, the doses to personnel are calculated in details and are optimised when needed. The gaseous effluents are evaluated within the material and radioactivity flow control system and compared with the on-site authorised limits for gaseous effluents and optimised when needed by selecting proper dismantling techniques with various factors for releasing of aerosols. The discharged water is kept under the authorised limits for discharging.

7.2. Methods for calculation of the dose uptake

The principle of calculation of dose uptake during decommissioning activities, as applied in the computer code OMEGA, is the determination of manpower components for the individual professions (trades) of the work group in the first step, and calculation of the exposure of individual professions for these manpower components for "elementary activities" as follows:

- Calculation of manpower for elementary decommissioning activities
- Distribution of calculated manpower to individual professions of the working group
- Extending the manpower for non-productive working time components
- Calculation of external and internal exposure based on local radiological conditions, protective means and manpower components which represents the duration of the elementary manpower component

Calculation of manpower and exposure is different for different groups involved in decommissioning activities. There are three main types of decommissioning activities regarding the exposure of personnel:

- Decontamination and dismantling. The exposure is dominated by dose rates at a defined working distance to the equipment to be dismantled. The exposure can be controlled by: controlling the duration of stay of workers in the vicinity of the dismantled equipment, by application of more personnel or by application of remote dismantling or by pre-dismantlement decontamination of equipment.
- Work at facilities for radioactive waste processing. The exposure is controlled by appropriate means at the work place (e.g. with shielding). The exposure is kept below the annual exposure limits.
- Period-dependent activities like surveillance, maintenance, management, technical support. The exposure is controlled by management of the working time including the management of dose budget which is kept below the annual exposure limits.

Critical decommissioning activities from the point of view of calculation of exposure are the dismantling activities, where the personnel, carrying out the decommissioning activity, are located close to the dismantled contaminated equipment. The dose uptake during decommissioning activities is calculated as external exposure and internal exposure. Eternal exposure is calculated based on duration of activities and dose rates at working places and the internal exposure based on concentration of aerosols generated during decommissioning activities. Conversion factors of individual radio-nuclides, personnel protection means applied and breathing data are taken into account. The computer code evaluates the possible internal exposure without any personnel protection means, and according to the evaluated value, the code allocates the relevant protection which is then used in calculation of internal exposure.

The “bottom-up” principle implemented in the OMEGA code enables one to calculate the exposure data on the level of an elementary decommissioning activity. Each hands-on decommissioning activity is decomposed into elementary productive and non-productive manpower components and for each manpower component, the relevant radiological data are allocated based on inventory data of the facility. The model time structure of an elementary decommissioning activity, as defined in the calculation model, is presented in Figure 11. The selected approach is an attempt to model the real sequence and content of decommissioning activities of this type.

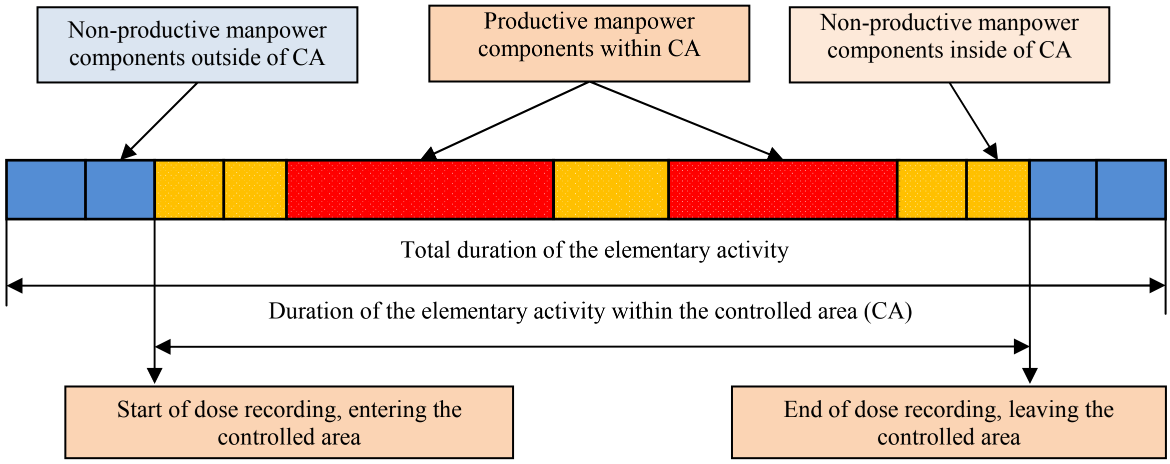

Fig. 11. Model working time structure of an elementary decommissioning activity.

The first step is calculation of productive manpower components. All non-productive manpower components are calculated based on coefficients which increase the basic productive manpower. The productive manpower components, depending on working conditions, can be increased due to differences between the ideal and local radiological.

Another aspect which is taken into account when calculating the exposure of workers, is the different working conditions for different professions of the working group. For each decommissioning activity, a working group is defined - professions needed and number of workers per professions. Various professions of the working group are exposed differently from the contaminated equipment, from the average dose rates in the rooms. In the case of internal exposure the OMEGA code allocates a personal protection means, depending on local conditions, in order to decrease the amount of inhaled aerosols. Therefore the exposure of personnel is calculated on the level of the individual professions of the working group.

In summary the principal scheme for calculation of exposure of personnel is presented in Figure 12. The main input data needed for calculation of exposure of personnel are the following:

- Manpower components resolved according to individual professions of the working group
- Composition of the working group – professions and number of workers per profession
- Dose rate data – dose rate 0.5 m from the equipment (average working distance), average dose rate in individual rooms of the facility, average background dose rate of the facility
- Concentration of aerosols in working places, depending on release factors of cutting techniques, local and facility ventilation, personnel protection means allocated

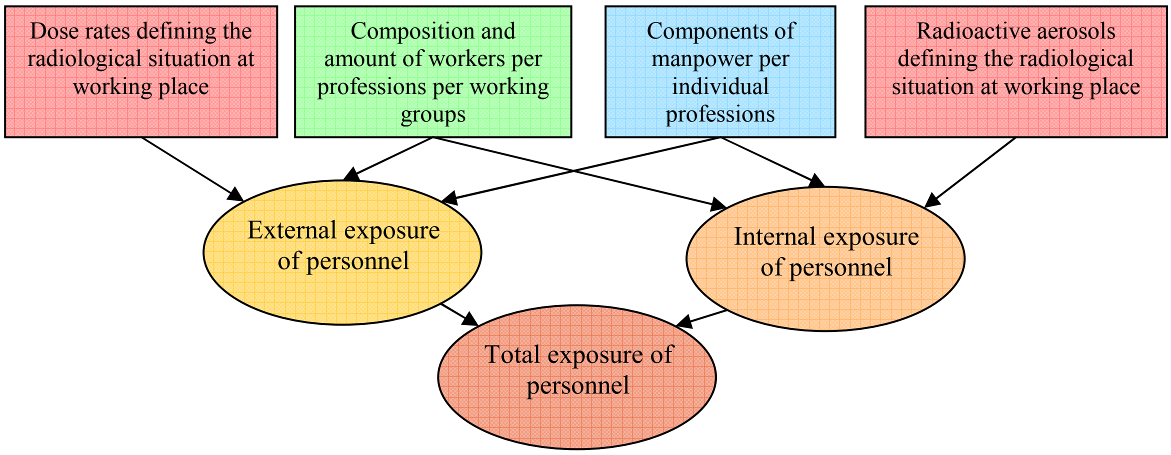

Fig.12. Principal scheme for calculation of exposure of personnel.

7.3. Optimisation of calculation of dose for personnel

The individual professions of the working group are exposed in different way in accordance with the type of work they perform. The most exposed professions are those who directly perform the dismantling and are most exposed to the dose rate of the dismantled equipment. For other professions the average dose in the room is dominant. For the rest of the working time, the dose rate in the background of the controlled zone is applied. These conditions are used in calculation of the dose uptake for individual professions of the working group expressed by coefficients of effective stay in the working distance from the equipment and in the average dose rate in the room.

In the conservative approach, the dose rate in the room is used for dismantling of all items in the room. The methodology developed for the OMEGA code calculates the dose items, relevant for the room. This is done more realistically by taking into account the decrease of the average dose rate in the room during dismantling of the equipment in the room. The method corresponds to application of the ALARA principle where the equipment with the highest dose rate is dismantled first in order to decrease the resulting average dose rate in the room. The effect is presented in Figure 13 where the conservative and optimised approach is compared.

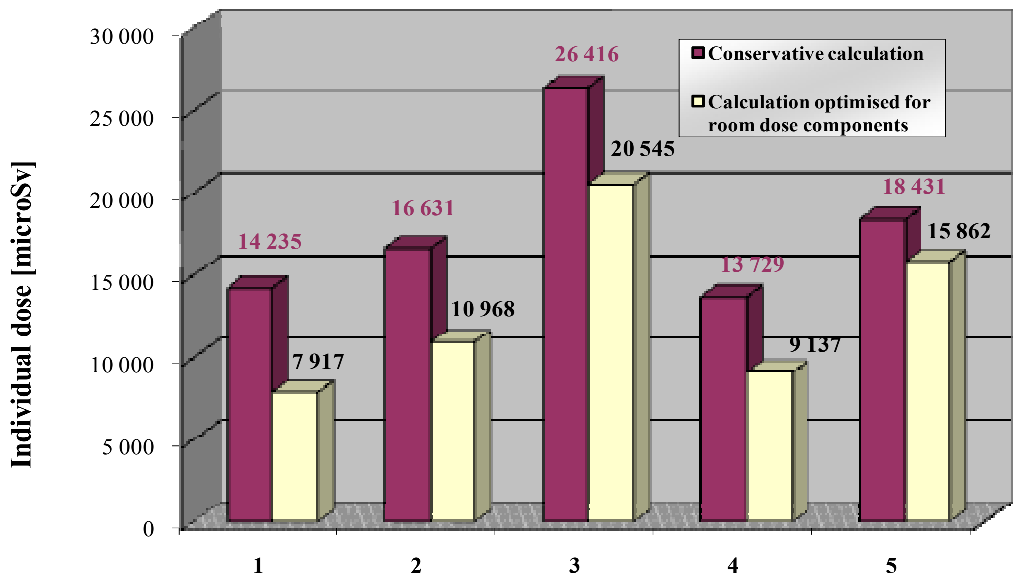

Fig. 13. Application of optimized calculation of the dose uptake during dismantling taking into account decrease of the dose rate in the room during dismantling.

7.4. Methods for evaluation of dose to personnel

The sum of the external and internal dose for individual professions of the working group is summed and presented as total collective dose for individual professions. After summing over all professions of the working group it is presented as the overall dose for the discrete elementary decommissioning activity. The summation upwards is done through individual phase up to the overall value of collective effective dose for the project which is used as one on the analytical parameters for the multi-attribute analysis for selecting the optimal scenario of decommissioning.

According to the general procedures for protection of workers, the dose for individual workers should be evaluated and controlled according to the ALARA principle and in any case should be lower then the annual limit of 20 mSv per individual.

The bottom-up principle and profession-resolved approach implemented in calculation of decommissioning parameters in the OMEGA code, enables one to evaluate analytically the dose to individuals performing the discrete elementary decommissioning activities. By summing the data over the duration of the given decommissioning phase, realised by one working group and comparing the duration of the phase with one year duration, it is possible to evaluate whether the annual limit of 20 mSv per year per individuals would be met within the evaluated dismantling phase.

Depending on duration of the decommissioning project or its phases under evaluation, the distribution of individual effective dose can be managed by following measures:

- Application of pre-dismantling decontamination
- Involving more identical working groups
- Managing the performance of decommissioning activities by mixing the activities performed under higher exposure risk with the activities with lower exposure risk
- Implementation of remote controlled operations (ratio cost vs. “saved Sv’s” can be evaluated)
- Deferring the dismantling. The time point, when the individual dose in under the annual limit for all profession can be found and phases of deferred dismantling can be justified analytically

A new approach was developed for visualisation of exposure risk properties of the evaluated decommissioning phase for individuals of the professions of the working groups. It is based on construction of the spectrum of manpower components (vertical axis) for an individual realised in the given dose rate interval (horizontal axis). The dose rate, in which the elementary decommissioning activity is being performed is calculated as the ratio of the sum of calculated dose components corresponding to productive and non-productive manpower component of the individual (Fig.11) and the total duration of the elementary activity within the controlled zone (dose rate normalised to total duration within the controlled area). This procedure is an analogue to real recording of the dose to personnel when entering and leaving the controlled zone (Fig.11). The method can be used for detailed evaluation of critical dismantling phases, to determine whether the annual limit of the dose limits will be met.

The example of such manpower spectrum and corresponding individual effective dose spectrum is presented in Figure 14. For comparison, the individual effective dose spectrum is presented for two starting points, the second 15 years later than the first. The dominant radio-nuclide in this model case is Co-60. In comparison with presenting the total values, this approach keeps the identity of individual decommissioning activities constant in presenting the results.

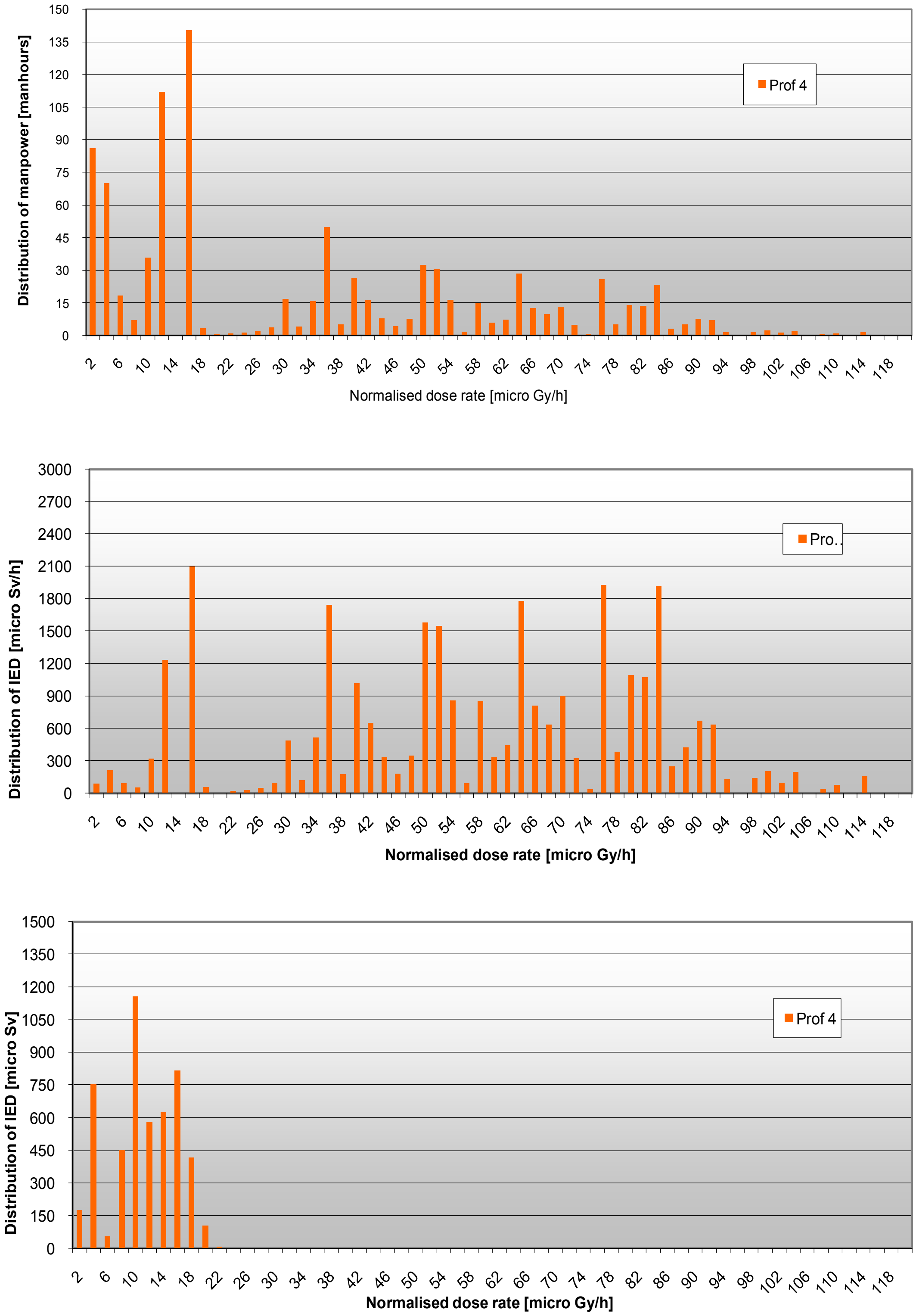

Fig. 14. Manpower spectrum for one individual of the most exposed profession of the working group for dismantling of one selected system; corresponding individual effective dose (IED) spectrum and the same IED spectrum for dismantling 15 years later (from top to down).

8. Optimisation of waste management scenarios

8.1. The approach to waste management in the OMEGA code

Special task in optimisation of the decommissioning option is the optimisation of decommissioning waste management. The implemented tools enable one to create a multi-stream waste management system which can cover several periods of waste management linked to individual decommissioning phases. The waste streams are pre-defined in the form of waste scenarios. The user can include the selected scenario into the calculation option. The principle scheme of waste management in the OMEGA code is presented in Figure 15.

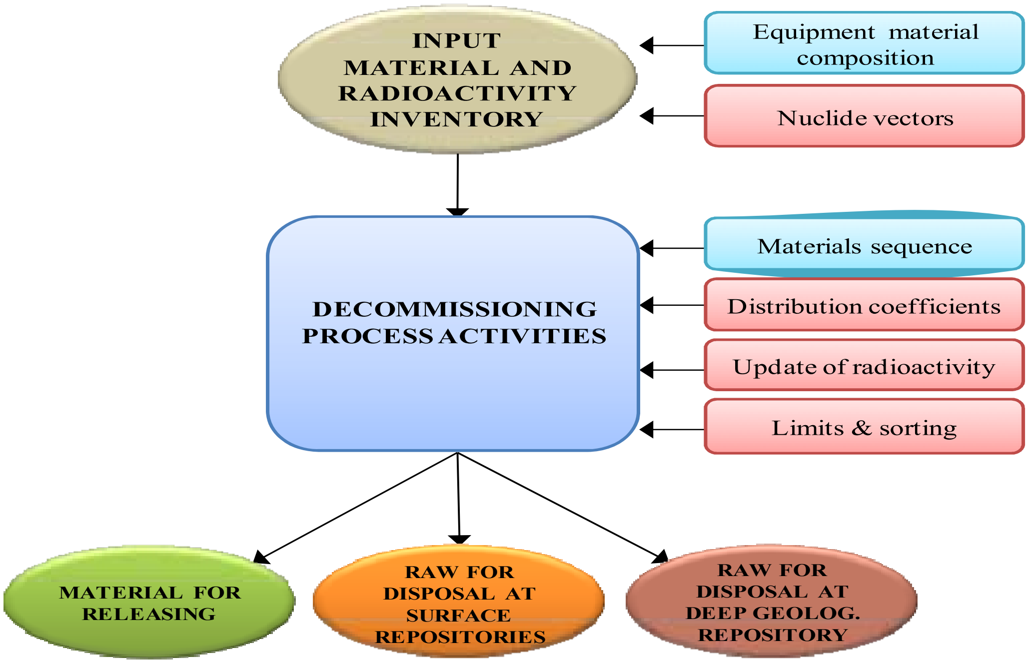

Fig.15. Principle scheme of waste management in the OMEGA code.

The amount of wastes entering the individual streams is the result of calculation of relevant waste generating items like dismantling, decontamination (wet and dry), demolition and this on-line data link increase the accuracy of calculation of the parameters for waste management. The secondary waste items are calculated based on distribution coefficients. Keyboard data entry of waste items is feasible for calculating the parameters for management of operational waste (not the result of previous calculation).

The range of on-line waste management activities includes all waste handling activities from the origin of the waste (or the keyboard entry) up to the release of materials or disposal of conditioned waste at surface or geological repository.

Complex schemes were developed to cover the management of waste from overall decommissioning process. The main types of waste managed within OMEGA are:

- Metal waste from dismantling - stainless / carbon steel, colour metals, electrical cables
- Non-metallic wastes from dismantling - thermal insulation materials, combustible waste, compactable waste

- Special materials (graphite)
- Waste from decontamination / dismantling of building structures
- Liquid waste from wet decontamination processes, from waste treatment, sanitary systems
- Special interim products of waste treatment
- Protective clothing and other personal protection elements
- Contaminated soils
- Non-contaminated materials from dismantling outside of the controlled area and from demolition

8.2. Waste management techniques

Extent of waste management processes for waste from waste generation decommissioning procedures and waste defined by the user keyboard entry is the following:

- Fragmentation - manual or remote controlled, selected automatically based on radiological status, defined in the inventory database and actualised for the time of execution
- Sorting of materials - takes into account the nuclide resolved limits for unconditional release, acceptance limits for surface disposal and acceptance, and limits for individual processing technologies. The sorting procedures define the amount of waste in individual waste stream
- Processing of steels - scenarios consider the decontamination (wet or dry) and melting as technologies (individually or combined) for maximising the release of steels
- Electrical cables - streams with decontamination, fragmentation and treatment techniques identified for release and for final conditioning
- Liquid waste / sorbents - evaporation, bituminisation, cementation, vitrification
- Incineration, compacting, cementation into drums and release are considered for other waste types
- Cementation of treated waste into cubical concrete overpacks for disposal, radiation monitoring of overpacks, transport to disposal and disposal of the overpacks
- Releasing of materials - radiation monitoring of drums, ingots from melting, discharged waters
- Storing of fragmented materials in drums and (radioactive decay over time)
- Landfill of contaminated soil on the nuclear site and processing of non-contaminated materials - recycling of concrete, back-filling

Other technologies could be implemented in new modules.

The review scheme of waste management in the OMEGA code for decommissioning costing and for general waste management project-optimisation, including the keyboard data entry, is presented in Figure 16.

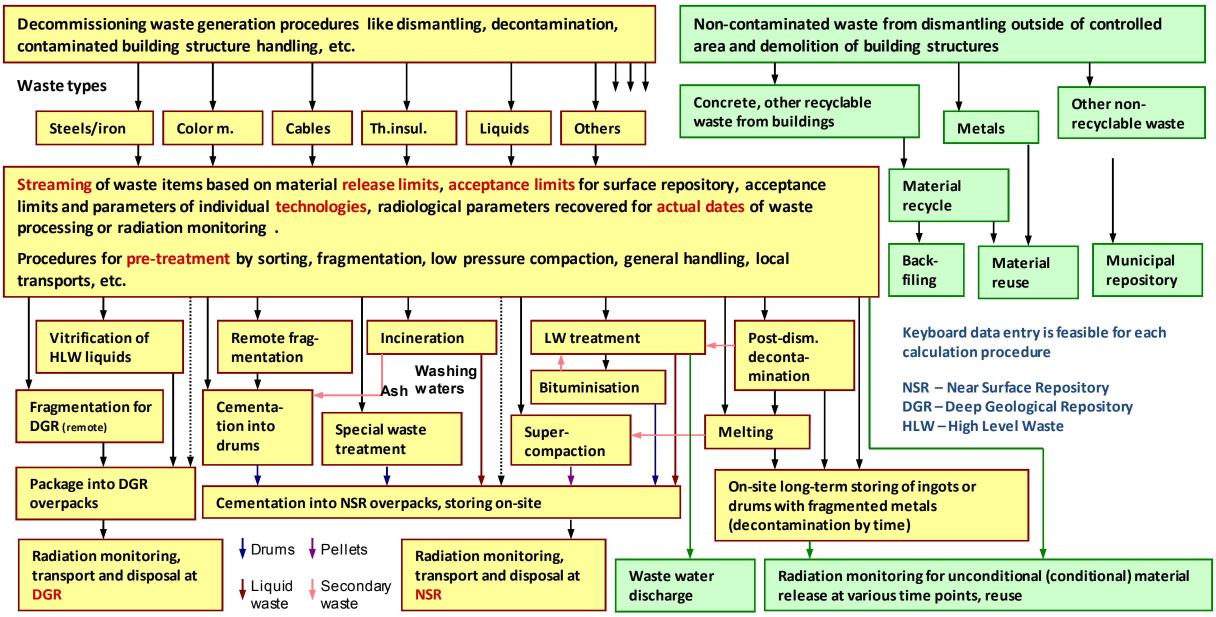

Fig.16. Review scheme of waste management in the OMEGA code.

8.3. Examples of application of the waste management of the OMEGA code

The properties of the system for material and radioactivity flow control enable one to perform various parametric analyses. Examples are presented via two model cases for evaluation of the amount of steel in various waste scenarios. In Figure 17 is presented the model case for analysis of the effect of the pre-dismantling decontamination and in Figure 18 the model case analysis of deferring the dismantling.

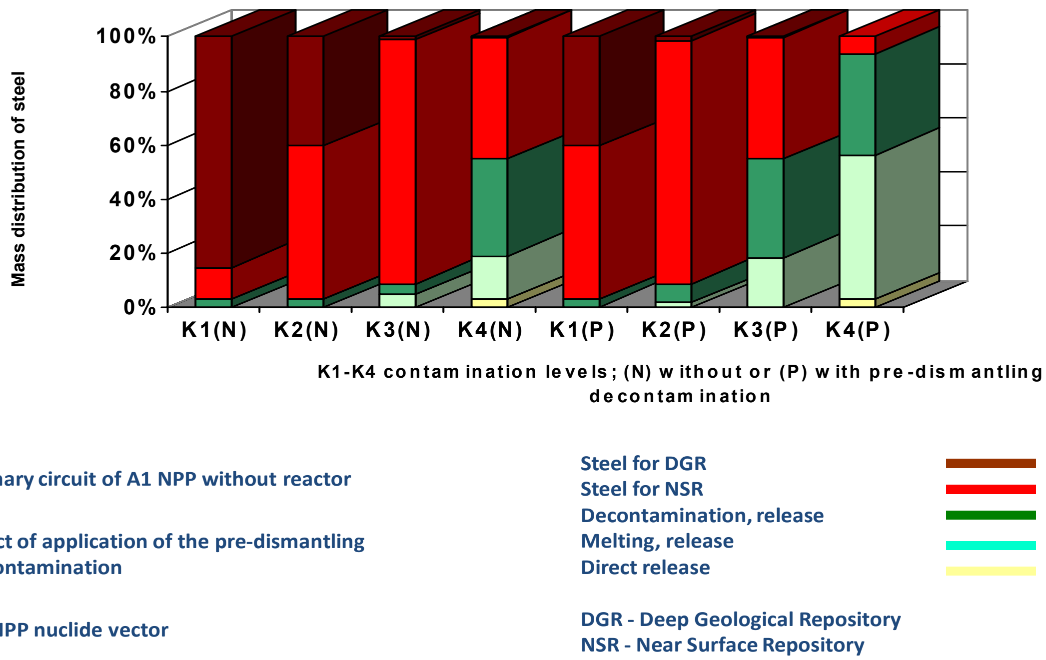

Fig. 17. Model case for the analysis of the effect of the pre-dismantling decontamination.

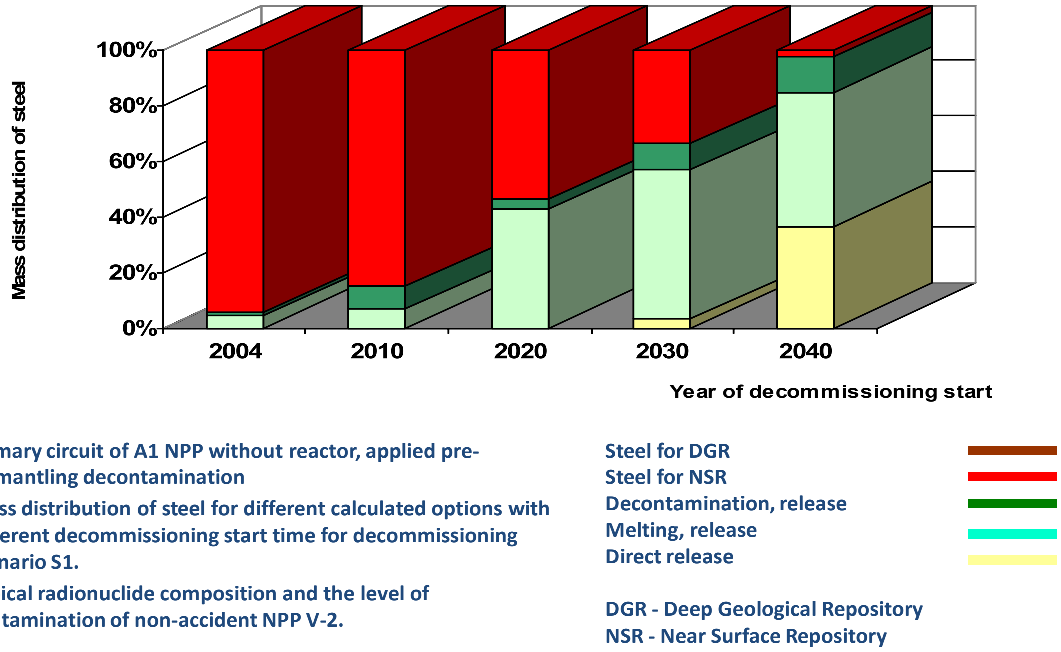

Fig. 18. Model case for the analysis of deferring the dismantling.

8.4. Example of application of the waste management system conditional release of metals

The concept of material and radioactivity flow control in decommissioning was applied for evaluation of the conditional release of carbon steel from decommissioning under the following assumptions:

- The conditionally-released metals will be used for constructing of rails
- The limits for conditionally releasing the metals is that the dose during construction of the rails and from the target application will not exceed the annual limit for public of 10 µSv/year
- A use will be found for the metal waste containing the short lived nuclides (Ag-110m, Co-60, Mn-54, Zn-65)
- The final location of the rails will be outside urban areas, on lines with lower traffic, where the replacement of rail is planned on a long term scale, in the model case only after 50 years

The scenarios for evaluation of the dose to public include the assembling of the rail sets (25 m length), installing the rails on site, welding of the rails, periodic maintenance of rails and passengers travelling daily for 2 hours. The procedure for evaluation of conditional release was the following:

- Developing model scenarios in Visiplan software (developed at SCK/CEN) in order to define the maximal content of radioactivity (Bq/g or Bq/cm2) in steel of the rails for the level 10 µSv/year (one example is presented on Figure19)
- Evaluating the amount of carbon steel for limits for unconditional release based on the current value of 0.3 Bq/cm2
- Evaluating the amount of carbon steel using a limit for unconditional release of 13,7 Bq/cm2, as defined by modelling in Visiplan for various scenarios of waste management (i.e. through application of pre-dismantling and post-dismantling decontamination)

- Aligning the individual steel items according to the inventory database (Fig.20) with the value for unconditional release, and the summing up of the items to the level of conditional release (the OMEGA code links the calculated data with the individual items of the inventory database)

Fig. 19. Example of modelling - installing the rails on site and the model in Visiplan software.

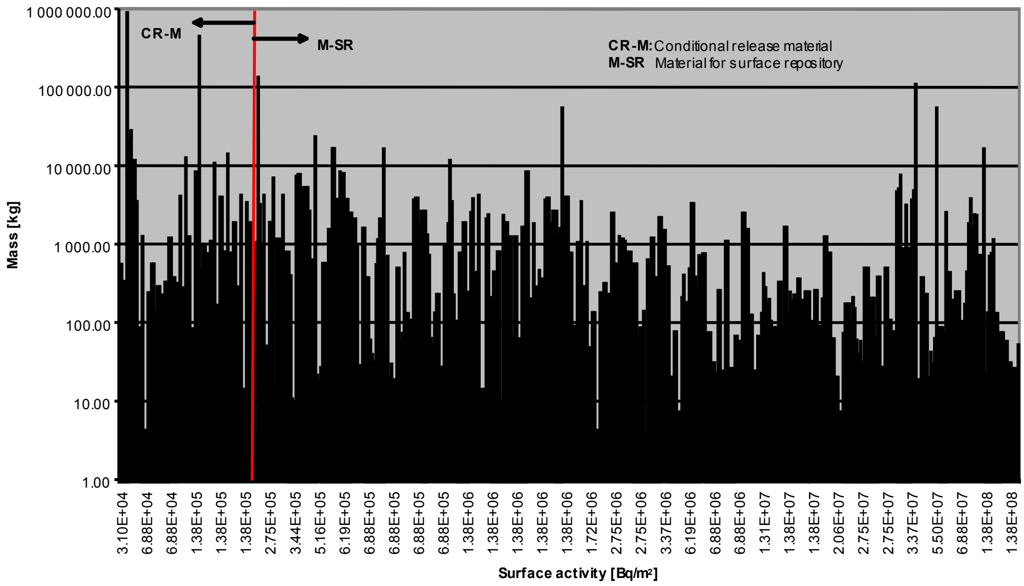

Fig. 20. Spectrum of carbon steel inventory items aligned by specific surface contamination.

As a result, 3 260 to 4 340 ton of carbon steel can be conditionally released, depending on scenario of steel treatment analysed.

9. Interactions with CRP members

Discussions were held with other participants during the meeting of the Co-ordinated Research Project, at Halden, Keswick and Rez. The subjects of discussion were the techniques for dismantling & decontamination of systems and structures, demolition of structures and parameters of techniques. Examples of application of the techniques, their availability on the market, aspects of operation and maintenance, materials and media needed for operation were also discussed.

Other discussions were oriented to methods for evaluating decommissioning parameters like cost, exposure, manpower, waste issues and other parameters, methods of calculation, structure of data and presentation of the data. The structures and properties of software used for evaluation of decommissioning parameters were discussed.

The facility inventory databases (structure and data) were discussed, as were the methods for data collection and, procedures for developing the data especially when the radiological data are missing, such as modelling in Microshield software. Procedures for developing the inventory databases, and methods of managing and application of the inventory data were also discussed extensively with the other participants.

These discussions were effective and as a result many new ideas evolved related to evaluation of decommissioning parameters and related aspects. These included techniques for decontamination and dismantlement, waste management methods and data. Selected aspects and data were discussed in detail. No contracts with other participants were agreed and no additional visits in organisations of other participants were organised.

It can be stated that organisation of this type of projects by the IAEA moves forward the general knowledge in the project subject areas and creates new contacts. Projects of this type are excellent platforms for information exchange. The main author of this national contribution used the platform of this CRP for promotion of the standardised list of items for costing purposes [5].

10. Conclusion

The Coordinated Research Project, T2.40.07 “Innovative and Adaptive Technologies in Decommissioning of Nuclear Facilities” created an excellent platform for information exchange in selected topics in decommissioning. The goal of the subproject “Comparative analysis of decommissioning technologies based on model calculations and multi-attribute analysis of specific decommissioning cases of nuclear facilities” was to contribute to the planned overall objectives of the CRP in aspects of “Conducting cost-benefit or multi-attribute analyses of specific cases of technology comparison and selection” and of “Comparing innovative vs. adaptive technologies”.

The contributions of the sub-project are mainly the presentation of the newly-developed computer code OMEGA for applications in the decision making process in decommissioning and decommissioning planning. Main features of the code presented for application in these areas are the implementation of the standardised structure of items for decommissioning costing, the systems for on-line management of flow of materials and radioactivity in the decommissioning process and on-line optimisation of decommissioning calculation cases.

The implemented approach, from the point of view of practical use, is of one compact standardised calculation structure which involves all decommissioning activities as defined in [5] including the waste management. This compact calculation structure provides a complete set of data within one calculation run, including the decommissioning schedule in the form of a Gantt chart of the project.

The presented approach is facilitated by automatic generation of the calculation cases based on the templates of standardised calculation structure and the facility inventory database. In this way it is possible to generate, in a user friendly manner, a set of decommissioning calculation cases which are evaluated and optimised individually. This set of calculation cases provides a basis of objective numerical data for selection of the optimal decommissioning option using multi-attribute analysis.

This contribution also involves methods for optimisation of the application of dismantling techniques, methods for evaluation and optimisation of safety related parameters in decommissioning and methods for optimisation of waste management issues in decommissioning. Some new ideas are presented like evaluation of safety parameters related to personnel involved in decommissioning and the analytical evaluation of conditional release of metals.

Selected examples are presented for demonstrating the methods. The presented results are based on a set of model calculations performed in the frame of the project. Two facility inventory model databases were developed for these purposes, based on the real data of two NPP’s.

The platform of the CRP was also used for promotion of the standardised list of items for costing purposes [5].

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DECOMMISSIONING PLANNING FOR THE KIEV'S RESEARCH REACTOR WWR-M

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Abstract
The research reactor WWR-M of the Institute for Nuclear Research of the National Academy of Sciences of Ukraine (NASU) has been in operation for more than 47 years. Ukrainian legislation demands the decommissioning plan for nuclear installation at the operation stage. Initial decommissioning planning for WWR-M was performed in the framework of the Decommissioning Concept issued in 2001. Currently the Decommissioning Program (DP) has been elaborated. On the basis of selected decommissioning strategy, the DP determines and substantiates the principal technical and organizational measures on the preparation and implementation of the WWR-M decommissioning, the chain of decommissioning stages, the sequence of planned works and measures as well as the necessary conditions and infrastructure for its provision and safe implementation.

1. Introduction

The research reactor WWR-M of the Institute for Nuclear Research (INR) of the National Academy of Sciences of Ukraine (NASU) is one of the first research reactors constructed and commissioned in the former USSR. The main purpose of the reactor is the provision of neutron beams for investigations in different areas of physics and engineering. From the first years, the WWR-M research reactor constituted the scientific and technical basis for research not only of scientists of NASU but also other organizations in Ukraine and the former USSR.

The present technical condition of the reactor allows its safe operation no less than 8 – 10 years subject to an upgrading of some systems and elements. At the same time, in accordance with the present legislation, the decommissioning of the reactor must be considered by the operator as early as possible. The initial decommissioning planning was presented in the Decommissioning Concept issued in 2001 [1]. Further detailed elaboration of this document was performed recently in the Decommissioning Program (DP) [2]. Taking into account the peculiarities of the WWR-M reactor, the Decommissioning Program is directed towards the solution of the following tasks:

- comprehensive and timely planning of all decommissioning activities;
- use of modern methods for the management of all kinds of decommissioning activity;
- use of modern decommissioning technologies and technical tools;
- provision of safety norms, rules and standards for personnel protection;
- use of permanent systems for the collection, treatment and storage of information having a significant influence on the decommissioning process;
- accommodation of the gradual decrease of the personnel associated with the WWR-M reactor by means of phased implementation of works;
- minimization of the radwaste generation, its treatment and final disposal;
- incremental release of the reactor's site from ionizing irradiation sources, which are a subject of regulatory control, down to free release levels;
- provision of social protection for the reactor's personnel; and
- public information to clarify decommissioning issues with the aim of confirmation of safety measures, which are planned or carried out.

2. Design and layout of the reactor WWR-M

The reactor represents a heterogeneous water-moderated research reactor operating with thermal neutrons at a power of 10 MW_t, giving a maximum neutron flux of 1.5×10^{14} $n/cm^{-2}s^{-1}$ at the core centre (Fig.1). the reactor has nine horizontal experimental channels, a thermal column, and 13 vertical isotope channels in the beryllium reflector. It is possible to install 10-12 vertical channels in the core. The main reactor's characteristics are presented in Table 1.

Fig. 1. Common view of WWR-M reactor.

Table 1. Main reactor's characteristics

Reactor power	10 MW
Number of FA of the WWR-M2 type	262 (max), 156 (min)
Core volume	82 l
Maximal density of heat flow	490 kW/m^2
Water flow through primary circuit	1200 m^3/y
Water flow rate in the core	2.6 m^3/s
Water pressure at the core input	$1.35x10^5$ Pa
Pressure difference in the core	$1.5x10^5$ Pa
Maximal water temperature at the core outlet	50^oC
Maximal temperature of the fuel assemblies	95^oC
Maximal density of the thermal neutron flux:	- in core – 10^{14} n/cm^2s - near reflector (isotope channels) – 6×10^{13} n/cm^2s
Maximal density of the fast neutron flux (E > 0.8 MeV)	- at the bottom of hot cell – 4.8×10^{14} n/cm^2s - on supporting grate – 5.2×10^{12} n/cm^2s

The reactor is located inside a modern security fence at the KINR site. The reactor is 100 m above sea level. The KINR site, which employs about 1000 people including support staff, is administrated by NASU. The reactor building is shown in Figure 2.

Main reactor elements (system) are:

- reactor vessel (tank) with the core;
- experimental equipment;
- cooling circuits (primary and secondary);
- water purification system for primary circuit;
- control rod system and system for control of rector’s parameters;
- power supply system at the regular operation and in the case of disappearance of main power supply source;
- radiation protection system;
- radiation control and protection system;
- special sewerage system (collection, storage and treatment of liquid radwaste);
- storage for fresh nuclear fuel;
- temporary storage for spent nuclear fuel;
- emergency cooling system;
- special ventilation system and filtration system at the regular operation and in the case of accidents;
- reserve water supply system;
- fire-control system; and
- physical protection system.

Fig. 2. Reactor building.

The reactor vessel is made of AD1 aluminium alloy (Fe <0.3%; Si <0.35%; Mg <0.05%, balance Al) and is surrounded by concrete shielding. The vessel has an outer diameter of 2300 mm, a height of 5705 mm and a wall thickness of 16 mm. The reactor vessel has the volume of 22 m^3; the distilled water fills it and is the heat-carrier, moderator and biological shield simultaneously. Vertical section of reactor is shown in Figure 3.

Be-reflector consists of fixed and removable blocks. The fixed blocks have the following dimensions: diameter – 936 mm, height – 590 mm. It has the vertical holes with a diameter of 6 mm for cooling. Removable Be-blocks are of hexahedron shape (32 mm) with a length of 590 mm. The fixed Be-block has vertical and horizontal holes for the layout of experimental channels. Total weight of Be is 335 kg.

The biological shield of the reactor is mixed and consists of beryllium (150-200 mm), water (680 mm), cast iron (200 mm) and heavy concrete (2000 mm at whole reactor's height). Biological protection above the reactor core is carried out by the water layer (3500 mm) and cast iron cover (200 mm). The core is protected from the bottom by a water layer (1000 mm). The core is shown in Figure 4.

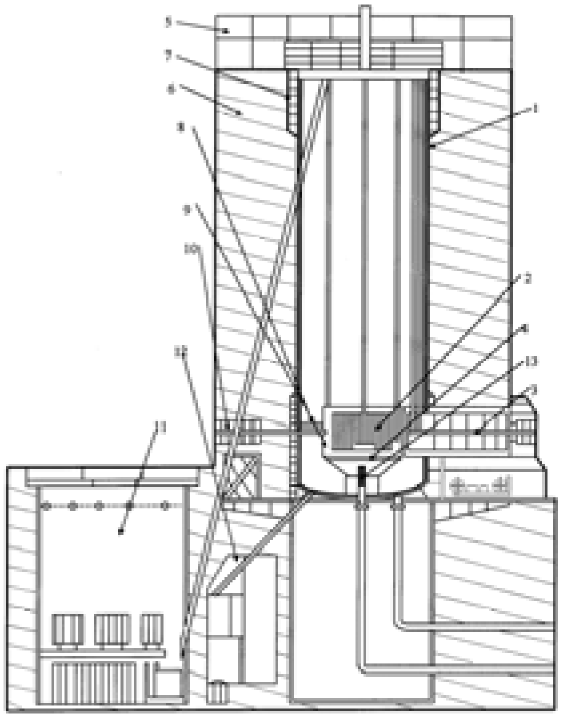

Fig. 3. Vertical section of reactor: 1 – reactor vessel; 2 – core; 3 – thermal column; 4 – core grids; 5 – reactor cover; 6 – biological shielding (heavy concrete); 7 – iron; 8 – grid for water flow; 9 – Be-reflector; 10 – horizontal experimental channel; 11 – SNF storage; 12 – "hot cell"; 13 – filter.

Fig. 4. View from above on the core.

3. Operational history and current status of reactor

The Kiev research reactor WWR-M was commissioned at the KINR site on February 12, 1960 and was used until 1994 mainly to study the radiation behavior of variety reactor materials. During this period it was operated about 100 hours per week to give an annual total of 3000-4000 hours. The total operating time was equal to 104500 hours (before 10.11.1993):

- capacity of 10 MW – 68.3%, 801.8 GWh
- capacity of 9.5 MW – 22.7%, 134.8 GWh
- transitional capacity – 9.0%, 43.4 GWh

The reactor was shutdown in 1993 and its core entirely off-loaded to the spent nuclear fuel (SNF) wet storage facility. Reactor operation was terminated in 1993 and until May of 1998 the reactor did not work. From May 1998 to the end of 2001 the reactor was operated according to an interim permit issued by the regulatory body. From May 2001 INR has had a permanent license for reactor operation.

During the period 1994-97, the systems for reactor control and fire protection were upgraded. In parallel with these improvements, a modern guard-post and a new security fence were installed around the reactor by the US MPC&A program. In this period the INR undertook numerous measures improving the research reactor nuclear and radiation safety, in particular:

- modern system of physical protection was commissioned;
- computer system for nuclear materials accountability was commissioned;
- new system for automated fire alarm was commissioned;
- two diesel power plants of 100-kW power each were installed and connected up (this is the source of emergency power supply system);
- lifetime of system equipment for research reactor control and protection was extended;
- lifetime of the research reactor tank, piping and primary circuit equipment was extended;
- operation of liquid RW processing facility was renewed; and
- lifetime of cables and switching units of safety-related systems was extended.

From May 1998 to the end of 2001, the WWR-M was operated according to the interim permit issued by the regulatory body. At the beginning of 2002 the Institute for Nuclear Research obtained the license for the research reactor operation. Term of the license expires with the end of the reactor operation. Lifetime of the vessel and primary circuit is not determined by design documentation. Negative changes beyond the design limit were not found as the result of inspections (hydraulic tests, ultrasonic, gamma-raying of welds). Present technical condition of the reactor allows its safe operation no less than 8 – 10 years in the case of an upgrading of some systems and elements. Now the upgrading works are in progress.

The National Academy of Sciences of Ukraine approved the «Strategic Plan for the use of research reactor WWR-M of the Institute for Nuclear Research» in July 2004 [3]. This multi-purpose strategic plan for the use of research reactor WWR-M is directed on the effective use of logically defined and analyzed actions on the nuclear installation. The plan's main aims are: (a) the co-ordination of works between the operator, researchers and users from different organizations; (b) determination of the user's needs and installation capabilities; and (c) provision of the reactor's stable operation by means of stepwise implementation of the planned strategic tasks. For conformity and co-ordination of the current strategic tasks, the strategic plan will be re-considered annually. The DP determines as the strategic goal the ***extension of the reactor operation until 2015.***

4. PREPARATION FOR DECOMMISSIONING

4.1. Legislative requirements

As a whole, the normative-legal basis of Ukraine is sufficient for decision regarding present-day tasks connected with the provision of safety and protection of the personnel, population and environment at the decommissioning of operating NPPs and RRs in Ukraine [4-6]. In this area, the normative-legal basis is corresponding to the international practice, takes into account the recommendations of IAEA, ICRP and other international organizations and satisfies the intentions of Ukraine for alignment with these.

The WWR-M research reactor was designed and constructed in the late fifties and decommissioning was not considered at the design stage. In accordance with the acting Ukrainian legislation and the IAEA recommendations, the decommissioning planning must be performed at the operation stage of nuclear installation as early as possible. The normative document "*General provisions on safety assurance of decommissioning of NPPs and research reactors*" [4] determines that the operator at various stages of the life cycle of nuclear installation should provide for the future decommissioning and to develop the ***strategy, concept and program*** (appropriate plans) of decommissioning.

At the operation stage of the installation, the operator should have the ***decommissioning concept*** agreed upon with the Regulatory Bodies. The decommissioning concept should be periodically revised taking into account new factors, which can influence decommissioning (facility state, development of decommissioning engineering means and technologies as well as radioactive waste management, requirements of legislative and normative documents, financial support, decommissioning experience, social aspects and others). This document, in a form as complete as possible, should reflect all aspects, consideration of which is possible at the given stage. Decommissioning concept should be periodically revised.

Preparation for the further decommissioning of Kiev's research reactor was started in the framework of the ***Decommissioning Concept*** issued in 2001 [1]. This document contains a common decommissioning approach and measures, which must be detailed and updated with the goal of preparation of the justified decommissioning plan.

Decommissioning Program [2] was developed in accordance with the requirements of licence EO No. 000051 for the activity "operation of nuclear installation", consisting of the research reactor WWR-M and spent fuel storage facility and it is the development of "Decommissioning Concept of research reactor WWR-M" (2001). The goal of the Decommissioning Program is to determine the main organizational and technical measures directed at the preparation for decommissioning and implementation of decommissioning, of the research reactor WWR-M.

4.2. Decommissioning strategy selection

Several peculiarities were considered at the decommissioning strategy selection for the WWR-M reactor [7, 8], namely:

- reactor is located in a large city of several million population and this increases the potential danger from possible accident after-effects and creates difficulties during the transport of SNF, radioactive waste and contaminated equipment;
- there are other active radiation-related installations on the institute's site (namely, electrostatic generator EGP and cyclotron U-240);

- activation of construction and elements located not far from the core is significant due to the high neutron flux density;
- decommissioning radwastes are differentiated from the operational ones by the wide variety and sizeable volumes;
- most of building construction is not radwaste. Only the surface levels in premises with the reactor's technological equipment will be radioactive waste;
- reactor has the horizontal experimental channels and other experimental installations, and their dismantling will be especially complex;
- in accordance with the Ukrainian law "On radioactive waste management" [9]:
    - article 3 forbids "*implementation of works on the radwaste disposal by the juridical and physical persons, which was generated during their activity, supplied and used the radioactive substances, nuclear installations*";
    - article 12 "*radwaste disposal is carried out only by the special enterprises for the radwaste management, which have a corresponding licence issued by established order*";
    - article 17 "*long-lived radwaste are a subject of disposal in solid form only to the stable geological formations with the obligatory transformation of them into the explosion-proof, flame-proof and nuclear safe condition, which guarantees the localization of radwaste within subsurface*"; and
- disposal facility for SNF and HLW in deep geological formations is absent in Ukraine and real terms of its creation are not determined yet.

Currently due to a number of reasons, it is impossible to plan for the future specific use of the site and reactor building. There are the isochronous cyclotron U-240 and electrostatic generator EG-10 on the INR site. The design lifetimes are not established for these research installations and, therefore, they will be in operation a long time even after the completion of decommissioning of research reactor WWR-M. A further operation of cyclotron and generator will need the continuation of a restrictive regime on the institute's site, independent of the state, to which the reactor's site will be transformed. Therefore, the most likely will be the use of the reactor's building with the "hot cells" as a separate laboratory for the development and application of radiation technologies [11]. The direction of utilization of this laboratory will be determined during the next few years taking into account the specific needs of the industry, in particular, the needs of nuclear power in the Ukraine.

It is necessary to mention that some time ago the possibility of constructing a new research reactor at the existing site near the WWR-M reactor was considered. In spite of obvious advantages of such a decision, the construction of a new reactor is entirely impossible today due to the stronger restrictions of current legislation concerning the location of new nuclear installations and, therefore, in the case of acceptance of decision for the construction, any new research reactor will be located outside Kiev.

Five variants of decommissioning strategy were considered at the outset, namely:

- ***immediate dismantling*** of the reactor;
- ***deferred dismantling*** with the safe storage of the reactor hall without partial preliminary dismantling;
- ***deferred dismantling*** with the safe storage of the reactor within the biological shield with partial preliminary dismantling;
- ***deferred dismantling*** with the safe storage of the whole reactor's building; and
- ***entombment.***

Entombment [10], as the cheapest alternative for dismantling and disposal outside the site, includes the drying of the reactor's pool and its filling with solid radioactive waste, for example, by the ***fragments*** of technological equipment or other radwaste after their immobilization in cement. This option is independent from the availability of disposal facilities beyond the bounds of the reactor's site and requires significantly smaller man-hours with minimal quantity of transportation. As the reactor building was not designed for long-term storage, this option will need additional protective barriers. The final state for this option is the passive safe monolithic structure, which will require long-term, but minimal control (restricted use). This decommissioning option is the cheapest one, but it is unacceptable for the population. Moreover, the creation of such storage facility can be unacceptable from the point of view of the environmental impact, specially in the case of an unforeseen external event or structural destruction resulting in the discharge of radioactive substances into the environment. In the future, if the regulatory policy is changing, this option will require significant expenditures for site remediation. As a whole, this option does not correspond to international practice, which requires an unrestricted release of the site after completion of decommissioning. Therefore, this option was recognized as unacceptable for the decommissioning of WWR-M reactor.

In the comparison of the three options of deferred dismantling and option of immediate dismantling, it is necessary to mark out the set of tasks which are common for all options, namely:

- reactor final shut-down and disconnection of control systems with the aim of prevention of the reactor repetitive start-up;
- spent fuel removal;
- removal of the process liquids and drainage of primary circuit;
- decontamination; and
- completion of removal and treatment of the operational radwaste.

Spent fuel removal is the obligatory prerequisite for all options since it is the requirement for the decommissioning licence issue. Subsequent implementation of decommissioning work has some differences, but without detailed assessment one can only say that the financial expenditures for all options are very close and, therefore, cannot be a determining factor at this stage in the selection of a decommissioning strategy.

For the selection of any option of deferred dismantling, it was assumed that it is possible to get some advantages due to the natural decay of radioactive substances and, thereby, to decrease the dose to the staff. In accordance with international experience, the expenditures for implementation of this option are practically the same as for the option of immediate dismantling. At the same time, in comparison to immediate dismantling, there are several disadvantages, namely:

- loss of operational personnel experience after long-term storage;
- necessity of maintenance of the corresponding document management system for decommissioning purposes;
- the option can be unacceptable for the stakeholders (especially to the residential population);
- additional expenditures will be necessary for maintenance during long-term storage; and
- potential danger of radioactivity release into the environment is remaining due to the accident or destruction of protective barriers.

In the case of the immediate dismantling option, the decommissioning process will rapidly reach its logical completion with the use of available personnel and, in that way, the transfer of a burden onto future generations will be absent. This is the only option acceptable for all stakeholders, first of all, for the local population.

In comparison with the arguments presented, it should be recognized that the financial and economical advantages are absent for all options, therefore, the decisive factors will be philosophical and ethical, as the technical complexity of the options is quite similar.

Consequently, for the WWR-M reactor, the option of immediate dismantling with removal of spent fuel and radwaste outside Kiev and return of the reactor's site for unrestricted use was selected. In accordance with preliminary estimations, the duration of decommissioning will not exceed 6 years. The Decommissioning Program foresees the strategy of immediate dismantling reasoning from the plans for further use of the site of the Institute for Nuclear Research of NASU. In accordance with the selected decommissioning strategy, the sequence of decommissioning stages was established as were the content of works and measures at various stages, including their duration as well as the necessary conditions and infrastructure for the timely and effective decommissioning execution. The final goal of decommissioning is the unrestricted or other re-use of the site.

4.3. Sequence of decommissioning stages

The Decommissioning Program establishes the following stages of decommissioning: ***termination of operation, final closure and dismantling***. Decommissioning of the reactor is preceded by the ***shut-down*** stage. The decommissioning license comes into force only after putting the facility into a nuclear safe condition, which means the absence of nuclear fuel on the site or its location within the site only inside the nuclear fuel storage facilities which provide long-term safe storage.

The ***shut-down stage*** is the final stage of operation ending with ***termination of operation***, which precedes the reactor's decommissioning. The main goal of activity at this stage is conversion of the reactor into a condition when the SNF is absent from the reactor's area or located within this area only in the SNF storage facility assigned for long-term SNF storage [12]. The final state of the reactor after completion of the shut-down stage is determined by the following: (a) SNF was removed from the installation; and (b) radioactive substances were located within the protective barriers as well as in the RAW storage facilities on the reactor area, or site, or transferred to disposal at the special enterprises for the RAW management. Duration of the shut-down stage must be equal to 3-4 years, such duration is determined mainly by the time of SNF cooling in the cooling pond (at least 3 years) and the time needed for further removal of SNF from the cooling pond.

The goal of the ***final closure stage*** is the transformation of the installation to the state that excludes the possibility of its use for the purpose for which it was constructed. The final state of the installation after completion of the final closure stage is characterized by the following: all systems and components which will not be used at successive decommissioning stages are dismantled, that exclude the possibility to use the reactor for the purpose for which it was constructed. Duration of the final closure stage must be about 1 year and depends on the technical preparedness for the implementation of works and measures mentioned above, as well as the availability of sufficient financial, material and staff resources.

The ***dismantling stage*** is the final stage of decommissioning and its goal is the same as the goal of decommissioning as a whole – achievement of conditions that reduce maximally any restriction on use of the site.

Currently, the following sequence of dismantling procedures is proposed:

- Dismantling of experimental installations at the horizontal channels (9 channels) (technology is available).
- Decontamination of the rooms in the controlled access area, which will not be used for dismantling purposes (technology is available).
- Removal of fuel assemblies from the cooling pond to the place of further storage (technology is available).
- Preparatory works for dismantling:
    - preparation and testing of tools, equipment etc.; and
    - preparation of thr area for segmentation.
- Dismantling of technological equipment at the reactor upper part:
    - dismantling the rotating cover plates;
    - dismantling of experimental channels;
    - dismantling of the elements of control rod system; and
    - dismantling of the bungs, pipelines and collimators.
- Dismantling of the first disk of thermal column.
- Dismantling of the reactor vessel:
    - dismantling of cover plates;
    - cutting-out of holes in the reactor vessel; and
    - removal and segmentation of vessel.
- Dismantling of ion-exchange and electrophoresis filters:
    - unloading of ion-exchange filter and barbotage of electrophoresis filter; and
    - dismantling of pipelines and both filters.
- Dismantling of primary circuit:
    - dismantling of attachment clips; and
    - dismantling of pipelines and equipment.
- Dismantling of main circulating pump:
    - dismantling of the pump frames and filters; and
    - dismantling of the embedded parts.
- Dismantling of the biological shield.

Final state after the dismantling stage is characterized by the following:

- there are no contaminated systems, elements and equipment in the reactor area;
- all decommissioning RAW are transferred for disposal to the special facilities for the RAW management; and
- the reactor's area is free from the radiation control required previously.

Another final state is possible when in the place of decommissioned installation, or alternatively a new nuclear installation, a facility for the RAW management or some general purpose facility must be constructed in accordance with such a project. Duration of the dismantling stage is about 2 years. The final decision about the duration of the dismantling stage must be determined and substantiated in the reactor's dismantling project and defined more exactly in the dismantling stage implementation program.

5. SPENT FUEL MANAGEMENT SYSTEM

The SNF storage volume enlargement is one of the problems of increasing the reactor's operational safety. Today, both layers of the storage are filled and there is a free place for the fuel assemblies in the core and reactor operation for one year.

The existing SNF management system at the reactor is providing the following functions: (a) the loading of spent fuel assemblies from the core to the cooling pond; (b) the temporary safe storage in the pond; (c) the provision of relevant chemical content of water in the pond and timely recharge of water; (d) the control of technological parameters in the pond; and (e) the reloading of spent fuel assemblies from the pond for the transportation and further treatment. This system is obsolete and cannot perform all functions entirely, namely, the pond volume is insufficient for storage, the requirements of new safety rules cannot be met at the SNF loading, and the possibility to carry out the preventive repair of the pond equipment in full is absent.

Reconstruction of the SNF management system foresees the elimination of the aforementioned disadvantages and will allow the following: (a) to increase the volume stored; (b) to improve the radiation safety at the SNF loading into the transport casks; and (c)inspection of the pond and, if necessary, to perform repair work. The proposed design of the new storage pond is shown on Figure 4. The new storage involves stainless steel vessel with the unit for the SNF assembly's storage in the form of aluminum alloy sections with holes for assemblies and sockets for rod-absorbers. The new vessel volume was designed for the storage of 1104 spent fuel assemblies (recalculated on single assemblies) and 228 rod-absorbers. The upper biological shield (involving special stainless steel and lead boxes) covers the vessel. The vessel is connected with reactor operation systems such as the ventilation, water supply, special sewerage and is equipped with control devices. In accordance with the design, the vessel is located in the tambour, which is connected with the reactor hall by a hermetic gate.

The reconstruction of the SNF management system includes:
- building the new cooling pond for SNF temporary storage;
- creation of the SNF reloading unit from the new storage to the transport casks; and
- perfection of the existing transport line from the old pond to the new one.

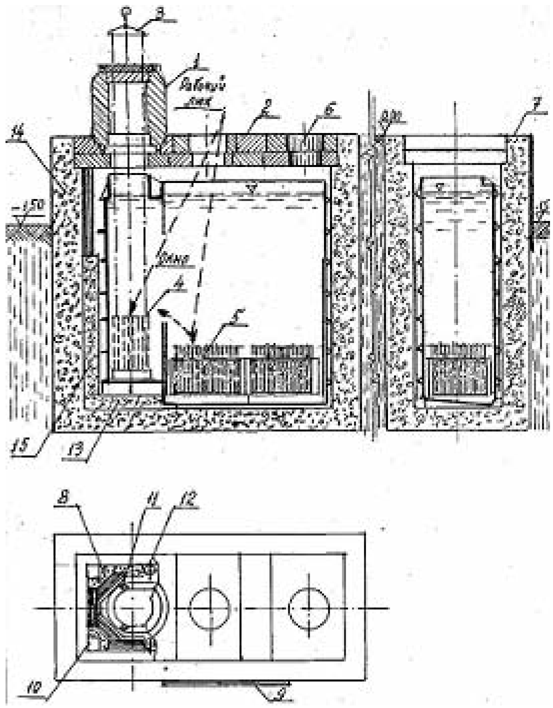

Fig. 4. Constructional scheme of CP-2 in the readiness for the SNF reloading into transport cask VPVR/M SKODA.

After commissioning of new storage in 2008, the reactor operation safety will be improved. Moreover, this provides the possibility to perform the inspection of old structures as well as to perfect the fuel assembly loading and facilitate SNF management for reactor decommissioning.

6. DECOMMISSIONING RADWASTE

Decommissioning radwaste will be different from operational waste in respect to variety and volumes. These radwaste can be classified in accordance with the radioactive contamination levels (high, intermediate and low activity), the physical conditions (solid, liquid, gaseous), and the treatment process (combustion, compaction, melting, etc.) [13]. The main part of the radwaste will consist of liquids and solids. The liquid radwaste will be low and intermediate activity, and the solid wastes will be of high, intermediate and low activity.

The following radwastes belong to the afore-mentioned categories:
Liquid radwaste:

- those generated directly at the dismantling operations (water flushing, dust suppression, gas cleaning) and decontamination solutions from the cleaning of dismantled segments of constructions and equipment before its further treatment;
- secondary liquid radwaste from the treatment of various radwaste; and
- dirty water from the sanitary sluice, waste from the laboratories.

The treatment and solidification of these radwaste are foreseen at the decommissioning.
Solid radwaste:

- main technological equipment (entire or segmented), including the reactor's elements, primary circuit pipe-lines etc;
- non-metal waste from the dismantled auxiliary equipment and pipe-lines;
- metal building construction materials after dismantling of premises;

- facing materials (sheet steel, elastron), plasterwork and broken concrete from the mechanical decontamination of premises;
- ventilation and process filters, filter cotton cloth, heat-insulation;
- concrete from the dismantling of biological shield and other premises; and
- construction and household rubbish, organic waste (special clothes, footwear, cleaning materials).

Metal waste from the dismantling and segmentation of equipment belong mainly to the 1st and 2nd categories. They are contaminated by the activated products of metal corrosion from contact with the heat carrier of the primary circuit. The largest problems with the non-metal waste are connected with the concrete from the dismantling of the biological shield.

It is planned to use an ***existing infrastructure*** for the collection, treatment and transportation of decommissioning radwaste. However, taking into account the large volumes and availability of large-scale elements, it is necessary to identify and/or develop appropriate technologies for radwaste fragmentation (including metal and concrete) as well as for the treatment of contaminated constructions (mainly metallic).

7. TECHNOLOGIES

7.1. Ranking of existing fragmentation processes

Metallic and concrete elements are subject to fragmentation. Since the fragmentation processes are determined by the desired external dimensions of fragmented elements, this factor will determine the selection of the fragmentation process. One can divide the fragmented elements into three categories:

- metallic elements;
- medium-sized concrete elements (having a thickness of less than 80 cm); and
- large-sized concrete elements (having a thickness of more than 80 cm).

The following criteria were considered at the ranking of existing processes:

- necessity of personnel access to high-radiation areas (applicability of remotely-operated tools);
- cost of the given process in comparison with other fragmentation processes;
- cutting rate;
- types of secondary waste, which are a result of fragmentation and have a potential danger from the radioactive contamination spread;
- safety:
  - radiation safety (simplicity of operation and control, ejection of aerosols and dust, etc.);
  - industrial safety (fire risk, temperature, explosive gases, etc.);
- reliability (the experience of application in the radiation-dangerous areas); and
- restrictions.

Taking into account the necessity for selection of the effective fragmentation processes corresponding to ALARA principles, the following criteria were accepted:

- minimization of the individual and collective doses and radioactive contamination during fragmentation. (this depends on the work time and dose rates); and
- minimization of cost.

In that way the most important criteria are:

(1) the amount of generated secondary waste important to the minimization of potential dangers and air contamination;
(2) applicability of the remotely-operated tools important to the minimization of individual and collective doses;

In general, this contributes to time limitation of the work execution (for the limitation of the corresponding irradiation time):

(3) cutting rate; and

The last criteria, influenced by cost-benefit relations:

(4) cost of the process in comparison with others.

7.2. Ranking and preliminary selection of the decontamination methods

It is necessary to divide the materials which are subject to decontamination.

Decontamination of the concrete elements

In the case when it is necessary to clean the porous surfaces such as the concrete surface (with cracks and rust) the methods involving surface removal are more effective in comparison with the methods involving surface cleaning.

Decontamination of the metal elements

The methods involving surface removal have the same efficiency as the methods with surface cleaning in the case of the decontamination of the surface of metal elements. However, taking into account that choosing the same method is more convenient for both kinds of elements (metal and concrete), the method involving surface removal seems the most applicable.

Possible decontamination of the containers

The methods involving surface removal are unsuitable for the decontamination of containers since these methods may destroy the container's integrity. Therefore, the methods involving surface cleaning are suitable for the container's decontamination.

The ranking criteria are the following:

- efficiency of the method application for the decontamination of materials with differing dimensions (efficiency on clear or painted concrete and/or metal surface; possibility of application independent of the component sizes);
- simplicity in operation and removal of generated waste;
- applicability of different operating regimes;
- capability to avoid secondary contamination;
- availability of restrictions: removal of contamination from the cracks and gaps, possibility of secondary contamination, and treatment of secondary waste; and
- safety (radiation and industrial safety).

The most important criteria from the point of view of the ALARA principles are the following:

(1) Efficiency of the method of application for the cleaning of a given material;
(2) Possibility of use of the remote-handled tools;
(3) Simplicity of radwaste treatment and disposal;
(4) Safety; and
(5) Secondary radwaste treatment.

7.3. The dust-suppression methods

Dust-suppression is aimed at eliminating dust generated during fragmentation and decontamination. The following criteria are most important:

- efficiency of the method of application for the decontamination of materials of different kinds and dimensions (efficiency on the clear or painted concrete and/or metal surface and possibility of application independent of the component's sizes);
- possibility to avoid contamination by free particles;
- simplicity in operation and removal of generated waste;
- applicability of different operational regimes;
- imposition of restrictions: removal of contamination from the cracks and gaps; possibility of secondary contamination, and treatment of secondary waste; and
- safety (radiation and industrial safety).

The most important criteria from the point of view of ALARA principles are the following:

(1) Possibility to avoid contamination by free particles;
(2) Simplicity in the radwaste treatment and disposal;
(3) Possibility of use of remote-handled tools;
(4) Safety (as discussed in 7.1); and
(5) Secondary radwaste treatment.

8. Interactions with CRP members and others

The information and experience received during the CRP implementation were used in the DTI Project NSP/05 R73R74R82U34 "Decommissioning planning for MR and GAMMA research reactors at the Kurchatov Institute, Moscow" through direct involvement of INR representatives in the tasks and decisions at all stages. The Project was funded by the United Kingdom Department of Trade and Industry (UK DTI) as part of a Grant-in-Aid assistance programme announced in July 2000 by the UK Government to address nuclear problems in the Former Soviet Union (FSU) countries. The purpose of this work has been to develop the main decommissioning planning documents for MR and, subsequently the work will be extended to the GAMMA reactor [14]. This is considered the basic part of decommissioning planning for these research nuclear reactors located at the Kurchatov Institute, Moscow.

9. Conclusions

The present condition of the reactor allows its safe operation for no less than 8 – 10 years assuming an upgrading of some systems and elements. Extension of the reactor operation till 2015 is considered now as the strategic goal. In accordance with the applicable legislation, the decommissioning planning must be performed at the operating stage of a nuclear installation as early as possible. The decommissioning program for WWR-M has determined the immediate dismantling as the optimal decommissioning strategy and provided the basis for its safe, timely and cost-effective implementation.

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RESEARCH AND DEVELOPMENT ACTIVITIES IN SUPPORT OF THE DECOMMISSIONING OF WINDSCALE PILE 1 — CHARACTERIZATION STUDIES

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Abstract
Windscale Pile 1 was a plutonium producing, graphite moderated, air-cooled 'pile' built in the north west of UK and operating in the 1950's. In October 1957 the reactor core caught fire releasing radioactive contamination into the environment — a major nuclear accident for the developing UK nuclear industry. Pile 1 was promptly closed down (together with its sister plant, Pile 2). Attempts to defuel Pile 1 were only partially successful and approximately 15 tonnes of uranium metal fuel is estimated to remain within the reactor core. Since that time the pile has remained in a quiescent state under a regime of surveillance and maintenance. Work is now gathering apace to decommission the reactor but the present state of the core internals has remained largely unknown during the shutdown period. This lack of adequate characterization has caused hiatus to the forward programme for decommissioning. Hence the initial step prior to dismantling is to understand better the present status of the pile. There are safety related issues presenting a number of technical challenges that must be addressed to enable a safe and cost effective option to be selected for decommissioning. .Many of these challenges have been now resolved by research-based studies and characterization work. Intrusive inspection into the Fire Affected Zone (FAZ) of the pile has now been possible. This paper details the approaches taken to provide a better understanding of the condition of Pile 1. The various technical issues have now been resolved providing a firm basis for the future dismantling of the pile.

1. Introduction

A sectional view of Pile 1 is shown in Figure 1. The reactor core consists of a 2000 te graphite moderator/reflector comprising about 50,000 blocks of graphite keyed together using a system of graphite slats and tiles (Fig. 1) generating an interlocking structure of approximately 15x15 m in horizontal section and 7.5 m deep. In Figures 1 and 2, slug channels are fuel channels and flattening holes are isotope channels. The core is penetrated by 3444 horizontal fuel channels of 100 mm diameter and 977 isotope channels (for irradiation of selected isotopes such as Li-6), each of 44 mm diameter. Each fuel channel contained a stringer of 21 solid uranium metal fuel rods clad in finned aluminium, and resting on individual, linked graphite 'boats' for loading and discharge purposes. The fresh fuel was loaded from the charge face from an ascending platform containing a refuelling machine. Fuel discharge was effected by pushing out the irradiated fuel by the incoming train of fresh fuel until the spent fuel fell under gravity from the pile discharge face into a rail-mounted transfer skip contained in a water-filled duct (Fig. 2). Transfer skips were then towed remotely on the rails into a cooling pond for storage and graphite boat removal prior to subsequent reprocessing of the irradiated fuel. The pile was designed to operate at up to 180 MW_t power (no electricity was produced) and cooled from a bank of blowers that forced air through the pile into a collection plenum and then exhausted to atmosphere through a vertical stack of ~130 m height via filters. The graphite core is surrounded by steel thermal shield plates, insulation boxes and a core restraint girder system - all encased in a reinforced concrete bioshield.

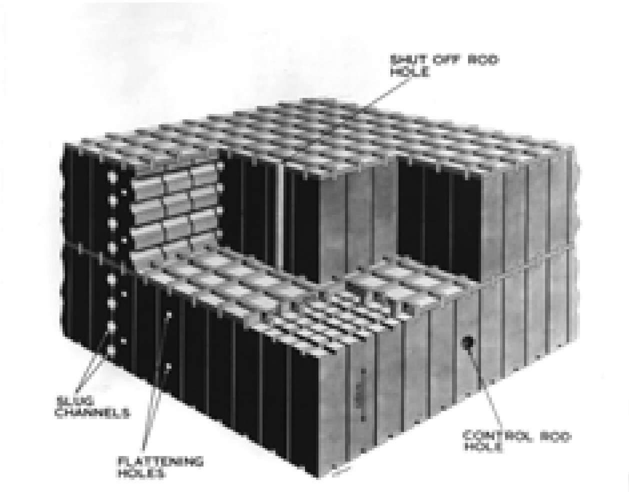

Fig. 1. Graphite block arrangement in Pile1.

Fig. 2. Cross-section through Pile 1 structure.

2. Pile operations

Design work on the piles started in the late 1940s. Pile 1 started to operate in 1950 as the principal provider of strategic defence materials for the UK nuclear deterrent. Subsequently, the second pile, Pile 2, was completed and started to operate in 1951.

The nuclear properties of the various materials used to construct the piles were not fully understood at the time of design and, in particular, the need to accommodate changes in graphite block dimensions caused by neutron irradiation (known as the Wigner effect) at the low irradiation-temperatures encountered (20 to 153^{o}C mean graphite temperature). Accordingly, the graphite core was designed with small gaps between the core blocks, back-to-front and side-to-side whose size varied according to core position. The Wigner effect manifests itself as ‘stored energy’ within the graphite crystal structure caused by defects/dislocations giving rise to dimensional changes in the graphite blocks. Hence, one of the operational tasks was to limit the stored energy by using nuclear heating to anneal the graphite core. Annealing, at least in part, removes the stored energy releasing additional heat and the graphite structure substantially recovers. During the working life of both piles, nuclear heating with reduced cooling air flow was used to elevate the core temperature to a point where temperature excursions occurred caused by the release of stored Wigner energy. It was during one such incident during October 1957 (the 9th anneal) that an uncontrollable temperature rise was experienced that led to a fire in the reactor core. The fire was eventually extinguished by a combination of water pumped into the core and by closing down the cooling air flow. The precise cause of the fire is still a source of conjecture. The sequence leading up to the fire, its ultimate control and post fire recovery, has been widely documented [1]. Not all the remaining fuel could be ejected from the core (neither by conventional nor by more energetic means) and up to 15 te may still remain. The use of water during the accident sequence has had consequences both for the pile in its current quiescent state and for its ultimate decommissioning. These consequences and the approaches taken to determine their safety implications are developed in the rest of this annex.

3. Technical issues related to dismantlement

The current research activities have been undertaken to support safety cases to modern standards for Pile 1 both, in its present quiescent state, i.e. pre-dismantling, and during the dismantling phase. Further, to enable a safe and cost effective option to be selected for future dismantling, the various safety issues must be addressed and solutions developed for their mitigation. The technical challenges to be addressed are discussed in [2] and summarized below:

- The remaining fuel mass and moderator is sufficient to present a potential criticality hazard during the Design Basis Accident (DBA);

- The graphite moderator was left in a partially annealed state following shutdown ––the quantity of Wigner energy within the graphite cannot be easily determined;

- The physical and chemical state of the fuel is not well understood and, due to the injection of water in 1957, the presence of pyrophoric[1] uranium hydride cannot be easily discounted. This material could be present in sealed 'pockets' that on exposure to air would oxidise exothermically. Hence, disturbance of the core either by a seismic event or during dismantling is considered to be a hazard, potentially leading to a thermal transient resulting in a release of Wigner energy and oxidation of core materials; and

- Damaged fuel and the accumulation of dusts and larger debris have been observed in the discharge exits to the horizontal fuel channels by CCTV survey (Figs. 3, 4). It is postulated that the levitation of graphite dust during a seismic disturbance could constitute an explosion hazard if ignited by a pyrophoric material, with potential pressurisation of the reactor containment and release of activity.

Fig. 3. Fuel element debris within FAZ fuel channel.

[1] spontaneous heating in the presence of an oxidant, normally air.

Fig. 4. Fire damaged fuel element displaced from its graphite boat.

4. Progress on resolving the technical issues

4.1 Criticality

The inventories of fuel and moderator in the Fire-Affected Zone (FAZ) exceed the minimum values required for criticality in an idealised maximum reactivity lattice. Past theoretical criticality assessments of Pile 1 have been unrealistically pessimistic, and have consequently indicated the sub-criticality margin to be small. The theoretical modelling work has been compromised by a lack of detailed knowledge of the remaining core contents and configuration. From analysis of the core reactivity measurements by direct measurements [3], it has been concluded that there is likely to be a substantial sub-criticality margin. Recent modelling of credible seismic disturbance scenarios for moderator, fuel and neutron absorbers has shown conclusively that the measured margin of sub-criticality would not be significantly changed. This has been supported by a sensitivity analysis of the effects of the numerous variables involved.

When considering the possible disturbance of fuel, isotope cartridges and moderator, it is assumed that due to graphite oxidation during the 1957 fire, the channels of the inner regions of the FAZ may have increased in diameter reducing the overall strength of the FAZ structure. Consequently, a likely outcome of a seismic shock could be fracture of the reduced graphite sections, with graphite fragments settling downwards, carrying fuel and isotope cartridges (as original or oxidised material), into a reduced vertical pitch array. It is pessimistically assumed that an increase in moderation could also occur due to the possible filling of cavities surrounding the fuel with graphite debris. The exact distribution and condition of the fuel and neutron absorbing materials is unknown, but the as-designed fuel array pitch and channel dimensions were very close to optimum for maximum reactivity. The analysis therefore assumes pessimistically that the horizontal pitch of the fuel remains unchanged.

The criticality safety assessment and the associated sensitivity studies have demonstrated that Pile 1 will remain sub-critical during current quiescent conditions and a seismic disturbance of the FAZ cannot be expected to cause a criticality.

4.2 Uranium hydride issues

During the 1957 fire, water was injected into Pile 1 in order to extinguish the fire and remove heat. The possibility that uranium hydride may have been formed as a consequence and could present a pyrophoricity hazard on exposure to air has been identified. Although the formation and survival of UH_3 in Pile 1 is considered to be extremely unlikely, its presence cannot be ruled out definitively in regions that may have been sealed since the post-accident clean-up phase. It has therefore been considered for many years that fuel should only be removed under an inert gas cover. A detailed assessment of the practicability of this strategy revealed the impracticality of its implementation, prompting a re-appraisal.

In earlier phases of the project it was considered that hydride in the core could be enclosed and protected from contact with air, and would exothermically oxidise if the enclosing material was mechanically disturbed. It was further pessimistically assumed that the inventory of hydride was sufficient to heat and ignite uranium in contact with it. Further contribution of heat was considered to arise from the release of Wigner energy and the ignition of isotope cartridges, leading eventually to self-sustaining graphite oxidation and potential for release of activity to the environment.

As part of this research, the formation and survival of uranium hydride in Pile 1 has been reassessed in detail. A principal argument used in the safety analysis is that the open conditions in Pile 1 with an air atmosphere were never conducive to formation of uranium hydride in the first place, even during water injection. If hydride did form in local transiently anaerobic conditions in 1957, it is unlikely that it will have survived the subsequent period of aerobic (oxidising) conditions. However, it has been assumed for the purpose of the safety argument that some hydride is currently present locally in Pile 1. A thermal model has been developed using the Fluent CFD code [4], tested and applied to conceptualised arrangements of fuel in a Pile 1 environment. The thermal model itself has several in-built pessimisms, and a sensitivity analysis has been carried out.

Under ideal conditions for propagation of a uranium hydride oxidation transient, with improbable hydride exposure and an impossibly concentrated inventory of hydride, it has been demonstrated that:

- Bulk uranium metal will not get heated enough to ignite or oxidise at a significant rate;
- The temperature increase at the graphite fuel channel wall will be so slight that neither graphite oxidation nor release of Wigner energy will be initiated;
- Isotope cartridges will not get heated enough to release radiological inventory beyond that which would arise from physical damage in a seismic event;
- There is no significant thermal interaction between neighbouring fuel channels; and
- Hydrogen generated from oxidation of uranium hydride cannot contribute to a thermal excursion promoting release of activity.

The overall conclusion is that even if uranium hydride is assumed to be present, improbably protected in anaerobic conditions, and its oxidation is stimulated by seismic disturbance, a thermal excursion causing a significant activity release will not develop. Any additional uranium oxides generated by oxidation of hydride alone will contribute an insignificant fraction to that already present and potentially rendered airborne.

The work described above was conducted to support the deferral strategy for Pile 1 decommissioning. During the latter parts of this CRP the work has been extended to an analysis of possible effects during fuel removal as a precursor to core dismantling. In this case, and noting the above arguments, it has been necessary to plan for 'low-energy' methods for damaged fuel removal from the channels. With this in mind, a fuel channel retrieval tool (FCRT) has been designed that uses a scoop (for fuel damaged substantially to debris) and a gripper system (for essentially intact fuel elements). The system has been trialled using a mock-up arrangement and has shown to be effective (Fig. 5). In support of the deployment of this equipment, the original computer modelling work has been extended to demonstrate that uncontrolled thermal effects (hydride oxidation, Wigner energy release) can be controlled adequately during fuel removal and subsequent waste management operations.

The formation and survival of other reactive compounds in addition to uranium hydride has also been considered, and it is concluded that other reactive materials will be in a form and or quantity that renders them insignificant.

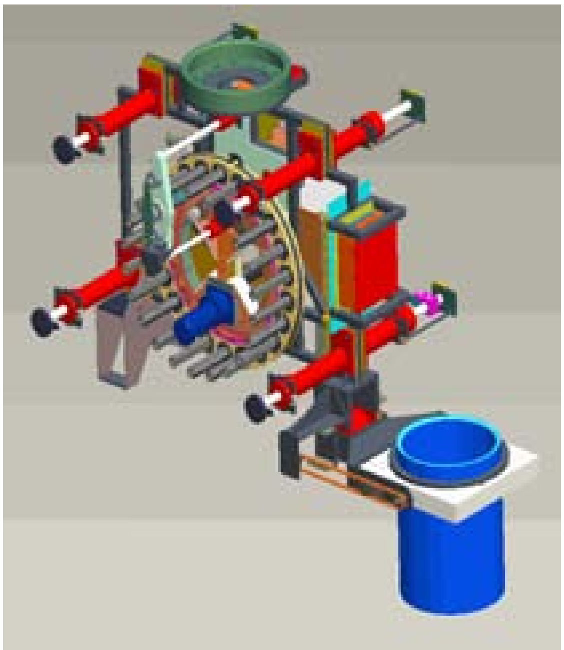

Fig. 5. Fuel Channel Removal Tool (FCRT) for fuel and isotope retrieval in Pile 1.

4.3 Graphite dust explosibility

The issue of graphite dust and its potential to cause an in-core explosion during decommissioning has recently received attention in the graphite decommissioning community [5]. Many countries operating graphite moderated reactors have carried out research programmes including UK, France, Italy and Japan. However, it has been noted, that since around 1890, no dust explosions in the graphite industry have ever been recorded. However, recent work, carried out as part of this research with pure nuclear grade graphite dust, has demonstrated that under ideal laboratory conditions it can be weakly explosible. In the safety analysis it is argued that for Pile 1 conditions a graphite dust explosion is highly improbable and can be dismissed. The experimental work on graphite dust explosibility was carried out using pure nuclear grade graphite. In reality the dusts observed to be present in the channels of Pile 1 will be a heterogeneous mixture, probably dominated by metallic oxides. It is generally established that inert components of any dust have the effect of suppressing its explosibility and the use of pure graphite therefore represents a worst case. Lead oxide[2], which is assumed pessimistically to be present, has an established catalytic effect, increasing graphite oxidation rate. Indicative tests however have not shown any observable effect, e.g. rendering the graphite dust more sensitive to ignition during an explosion scenario.

'Ignition' of airborne graphite dust suspensions requires a high energy power source. Uranium hydride oxidation has been suggested as a possible ignition source, but even assuming sufficient hydride in a highly reactive form was present, the reaction cannot provide the level of ignition energy and power input required. Electrostatic charge build up in a 'graphite environment' will be minimal as graphite is an electrical conductor. An energy pulse from a criticality could in theory contribute to dust ignition, but the criticality event has been dismissed. No credible dust explosion ignition source has therefore been identified.

It has been established that only the very smallest particles contribute to a graphite dust explosion, (less than 10 µm). Larger particles have the effect of a heat sink, quenching the reaction, and additional inert material, e.g. metallic oxides within the FAZ, act as suppressants. Even in laboratory conditions it has proved extremely difficult to produce the small particle size required, and there has been a persistent tendency for the fine graphite dusts to rapidly 'age', forming spherical agglomerations, giving an apparent reduction in reactivity. In view of this experience it is likely that an insufficient fraction of the dusts present in the core will be within the explosible size range.

A high concentration of airborne dust is required for an explosion to occur. The quantity of very fine graphite dust, free from inert material, required to produce an explosible airborne concentration in the fuel channels, and enclosed voids within the pile structure would be considerable. It is argued that the graphite damage needed to produce the necessary concentration of fines could not credibly occur.

5. Characterization processes

The resolution of the technical issues outlined above has been the direct result of a combination of characterization and desk-based studies.

At the pre-dismantling stage of Pile 1 it has been an important part of the characterization process to review and select technologies that could provide the most useful information on the present state of the core. The characterization process has been focussed on better determining the physical configuration of the reactor core internal structure post the 1957 accident and, in particular, the quantity and present status of the remaining fuel and isotopes. This process is somewhat unusual because for reactors that have closed down under normal operations the term 'characterization ' is generally used for determining the radiological condition in terms of activation and contamination fields and less so for the physical status. Although radiological characterization has been carried out using calculation codes for Pile 1 this has not been the main thrust of the present characterization activity — not least because the results of such calculations are compromised by the extensive contamination fields resulting from failed fuel during the 1957 accident.

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[2] Considerable quantities of lead were used in the core during operations to weight cartridges in order to prevent ejection by the cooling air flow

The various safety issues detailed in the bullets in section 3 have limited the scope for intrusive characterization and have thus hindered project progress. At the start of this CRP in 2004, it was not possible to carry out intrusive inspection within the FAZ of Pile 1. Hence, in the initial phases of this work it was necessary to adopt a rationale for selection of technologies that could be applicable in determining the present status of Pile 1 internals *by no intrusion into the FAZ*. During the subsequent course of the project, a much better understanding has been developed of the nature of the hazards culminating in an intrusive inspection of the pile during August 2007 (see Section 5.5).

5.1 Criticality calculations

The remaining fuel mass, isotope cartridges and graphite moderator within the pile is sufficient to present a potential criticality hazard. The physical state and precise quantity of the remaining fuel and isotopes is not yet fully understood. On the basis of non-intrusive inspection exercises carried out over many years on the charge and discharge faces of the pile, simplifying assumptions have been taken to account for the remaining quantities of intact fuel and isotope cartridges, partially oxidised (burnt) cartridges and residual fuel-bearing dusts.

The overall understanding has been compromised in recent years by the inability until recently to carry out detailed intrusive inspections outside of the FAZ region to determine the precise conditions in the reactor core. Accordingly, a dual approach has been used to determine the fissile content and hence the core reactivity. Firstly, models have been set up using the reactivity modelling code, MONK [6], based on the notional remaining fuel and isotope content assuming an intact graphite core structure. Secondly, an active neutron interrogation technique has been deployed based on the californium shuffler principle to assess the fissile content and hence estimate the remaining activity of the core by direct measurement [3].

The basis of the methodology applied has been to use a reactivity value obtained by the measurement, this is then used as a datum from which to assess changes to the core provided by the models. The assumption adopted is that the model is incapable of providing an absolute value of the core reactivity (k_{eff}) due to incomplete core knowledge, but is effective in estimating the changes in core reactivity (Δk_{eff}) that arise when making changes to the core configuration e.g. altering the fuel or moderator content.

Based on these assumptions, it has been demonstrated that the principal determinants of the core reactivity are (expectedly) fissile content but also the quantity of lithium containing isotope cartridges that still remain. These cartridges are very effective in suppressing the core reactivity due to thermal neutron absorption in Li-6. The reactivity assessments have demonstrated that the reactor could be up to 3% supercritical if these isotope cartridges are removed before the fuel. Such assessments indicate the need for a well planned rationale for reactor defuelling and isotope removal such that the core reactivity can be progressively lowered until a status of ‘criticality impossible’ can be attained i.e. irrespective of the sequence adopted for fuel and isotope removal no criticality becomes possible. Optioneering of the various removal sequences has been carried out to develop an optimum sequence thus minimising the handling and processing times for the retrieved materials. It may be necessary to utilise neutron poisons (e.g. boron) during the early stages of fuel removal to improve the safety margin.

5.2 Non-fissile waste inventories

Estimates of the inventory of radioactive wastes in Pile 1 have been estimated by neutron activation code calculation using the 1-D neutron transport code, ANISN [7]. A smeared (averaged) representation has been used to estimate graphite activation in the fuelled region of the core, through the thermal shield and into the concrete bioshield. This simple model has been used as the principal method of estimating waste quantities. However limitations exist in the approach since the radionuclides that are important for decommissioning have resulted largely from the activation of uncontrolled impurities in the bulk materials, e.g., Co in steels, N, Li in graphite and Li, Eu in concrete. Generic impurity levels were used in the models in the absence of firm estimates based on measurements (neither the importance of the impurities nor the technology to measure many of them at the trace amounts present were available during the 1950s when the pile was constructed). Another major limitation of reliance on standard modelling techniques is that, in the specific case of Pile 1, the core fire during 1957 has likely spread much of the volatile fission product content of the fuel around the core. Such unknown levels of contamination are not easily modelled and will be a significant contributor to the overall radioactive inventory.

5.3 Sampling

In order to improve on this situation, a programme of sampling is being conducted so that better estimates of the quantities of radioactive wastes in the various categories can be made. Initial sampling work has been centred on the bioshield concrete and the thermal shield plates separating the graphite core from the bioshield. Five concrete cores were taken from Pile 1 bioshield during 2006 — four on the west control face and one on the east control face of the reactor. A standard core drill designed for use in civil construction work was utilised together with flushing water to cool the drill bit and remove slurry via a wet vacuum cleaner. Secondary wastes were collected and drummed for disposal. Tenting was provided for containment and respirators used by the operatives.

Predictably, the core showing highest activity was located at the point of maximum neutron flux. Significant nuclides found to be present were H-3, C-14, Co-60 and Eu-152. The highest activity found was due to H-3 at the inner surface of the bioshield with typical values obtained by equilibrating with water for several days at ~170 Bq/g and total H-3 by pyrolysis at ~250 Bq/g. The first 1.8-1.9 m of the concrete bioshield in all cores was less than 4 Bq/g leachable tritium. As anticipated, elevated levels of Cs-137 were found as a result of the spread of fission products from the core fire - a sample of thermal shield plate steel recovered from coring had measured surface contamination levels on the face nearest the reactor of ~35,000 Bq cm^{-2} Cs-137 and ~500 Bq cm^{-2} Sr-90. A new campaign of graphite sampling will be designed to complement the existing data on the condition of the graphite from previous sampling work. Graphite samples will be removed from the Pile 1 core and analysed in order to provide information on the condition, density and waste categorisation. The information required will include dose-rate and radionuclide concentrations. Graphite samples from within the FAZ will also be analysed for mechanical strength, density and Wigner energy content in support of future core dismantling campaigns.

In addition to the extraction of core graphite samples, other core materials may be collected and analysed. Many fuel channels within the FAZ contain metallic oxide materials. Samples of these will be removed and taken for analysis in order to determine their characteristics. Since much of this material was produced from fuel and isotope cartridges oxidised during the 1957 fire it is important to discover the radionuclide content of the materials present both for waste management purposes and for the development of the decommissioning safety case. The volume of the FAZ is also yet to be accurately determined.

5.4 Visual inspections

A systematic review of characterization technologies indicated that TV/visual inspection methods would offer the best prospect for gaining the most information quickly on the presently unknown status of the reactor core. Such techniques should ideally:

(a) Identify anomalous features in the core structure:

- Loss of graphite during the 1957 fire that has resulted in reduced density/increased porosity via thermal oxidisation mechanisms or the presence of voids;
- Changes in the graphite structure which may have caused the core to slump e.g. identified by slippage of graphite blocks, slats and tiles and changes in the Wigner gaps (present between blocks to accommodate Wigner growth) — see Figure 2 for graphite core structure.

(b) Detect the presence and depth of any fuel and isotope channel blockages from the charge and discharge faces of the reactor.

(c) Identify the location and quantities of uranium metal fuel and isotope cartridges remaining in the FAZ.

(d) Identify and characterise the combustion and corrosion products which are present in the core post the fire (e.g. uranium dioxide, uranium hydride, aluminium oxide, oxidation products of the isotope cartridges).

In addition to the fuel and isotope channels that pass horizontally through Pile 1, there are a number of vertical full-depth penetrations over 15 m in depth. These penetrations were 'foil-holes' used for the introduction of flux monitoring foils and irradiation experiments during pile operations. The holes have provided a ready route for inspection of the internal condition of the pile.

Two phases of inspection work in the foil-holes have been carried out outside of the fire-damaged region. Initially, it was decided to adopt a prudent approach to the inspection by carrying out a non-intrusive survey (an optical inspection without entering the core) to prevent any possible disturbance to the core. Specialist viewing equipment has been developed and deployed successfully. The principle behind the design was to use a long focal length CCTV system to provide a non-intrusive view from the top of the reactor down the depth of the foil-hole, i.e. without entering the foil-hole.

A second inspection phase was carried out in June 2005 using an intrusive probe (a small CCTV) to confirm the earlier work (Fig. 6). Figure 7 shows a view at the base of a foil-hole indicating debris and the graphite in this region in overall good condition.

Fig. 6. Intrusive inspection equipment for foil hole examination in Pile 1.

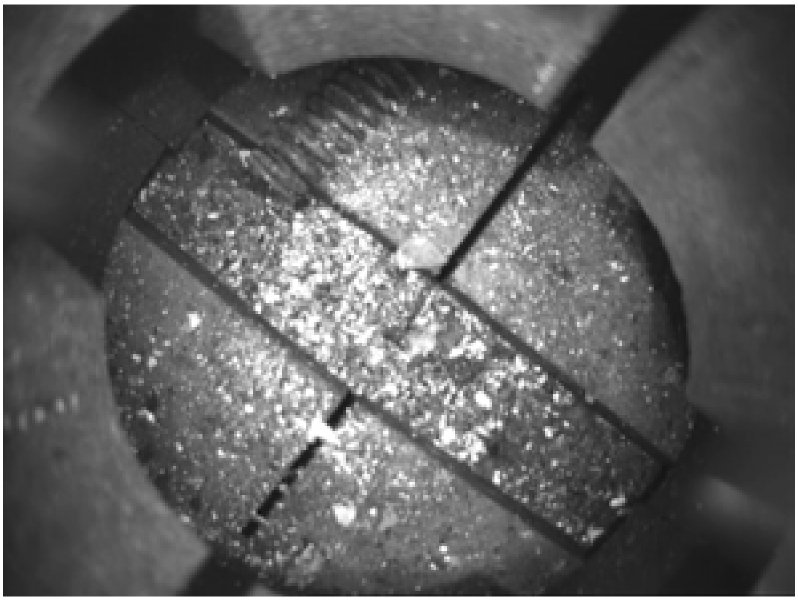

Fig. 7. View at the base of a foil hole showing Wigner gap between graphite block.

The systems have allowed excellent images of the pile internals to be recorded and have indicated that, nearly 50 years after a major incident, the graphite core brick structure is still in excellent alignment. Such information is of considerable value for the planning of future decommissioning operations.

5.5 FAZ endoscope inspections

A third phase of work has now been possible to extend the visual inspection work to the FAZ by horizontal insertion of a commercial flexible endoscope into the fire-affected fuel channels. A safety submission to support this work was accepted by the UK regulatory authorities on the basis of the desk-based work and earlier inspection work described in this paper. The initial phase of this work was started during August 2007.

The endoscope utilised a small diameter CCTV system and LED lighting on the end of a 14 m long flexible umbilical for direct insertion through the charge wall of Pile 1 into the FAZ (Figs. 8a, b). This internal inspection work has confirmed some of the earlier visual inspection work that had previously been carried out by views taken from the charge and discharge faces of the pile. A range of consequences of the 1957 fire are visible, from mechanical disturbance of the fuel stringers as the result of the post-fire removal exercises (Figs. 9a, b); melted fuel residues (Fig. 10) and oxidation damage to the graphite structure (Fig. 11).

Fig. 8a. Endoscope equipment for FAZ.

Fig. 8b. Interfacing the endoscope with Pile 1 charge face inspection in Pile 1.

Fig. 9a. Fuel channel 24/53 BL internals showing fuel element displaced from graphite boat.

Fig. 9b. Damaged graphite boats in channel 24/51 TR.

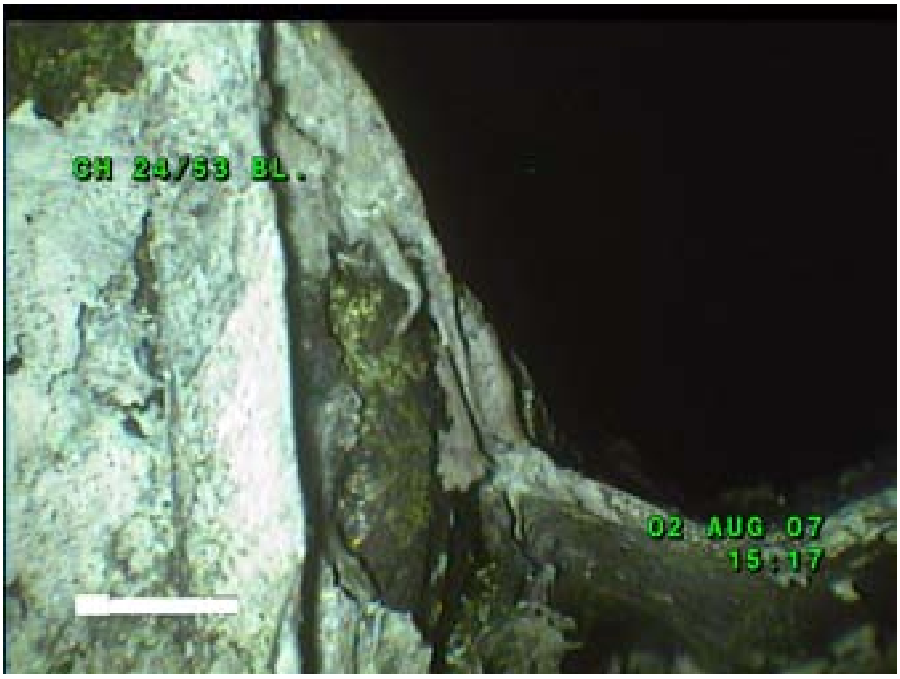

Fig. 10. Melted fuel in channel 24/53 BL.

Fig. 11. Oxidation damage in the graphite structure at channel 24/52 BL.

The extent of the damage detected will be used to ensure the design of the FCRT (Fig. 5) is sufficiently robust to successfully remove the fuel. Fuel removal is on the critical path activity for successful decommissioning of the reactor.

Whilst nearly every channel was a 'one-off' sample of conditions, conclusions from a preliminary review are:

(a) The FCRT is expected to be successful in removing accessible fuel for at least 95% of the fuel channels containing intact and damaged fuel; and

(b) At least 99% of the reactor graphite can be removed by the bulk removal methods (i.e. listing many blocks at once with a special removal assembly).

There is a damaged area estimated to be 8 charge pans (the fuel entry point) in total that contains about 5% of the damaged fuel and 1% of the graphite in the core. This area probably will require some additional tooling adjustments and/or equipment design revisions. This estimate will be revised as new information is available from further inspection of the remaining fuel channels.

The inspection during August 2007 represented the first time that the fire damaged region of the reactor core had been seen for 50 years since shutdown and has assisted in reducing the uncertainties in core condition that have been conjectured for many years.

6. Conclusions

The accident-damaged nature of the Pile 1 reactor core has led to a range of issues that would not normally be encountered for a reactor system that had shut down following normal operations. Hence, from the characterization perspective, this project has been unconventional since the early stages having focussed on the physical characterization of the damaged core.

Early studies have been compromised in the initial phases of this CRP by the inability to make intrusive inspections within the fire-damaged region of the core. A variety of safety related concerns have been identified that have resulted in hiatus in the characterization programme. However, desk top studies of the various issues related to criticality, possibility of uranium hydride present and graphite dust explosions have demonstrated successfully that such concerns have been based on very conservative assumptions not borne out in fact. This conclusion has enabled characterization work on Pile 1 to be progressed during the course of the CRP from an initial round of non-intrusive visual inspection into the undamaged sections of Pile 1 core through to intrusive inspection of the fire-damaged core. The results of this work have underpinned the methodologies for removal of the remaining fuel and isotopes in the core of Pile 1 since these constitute the major hazard. Further characterization work has now been started to quantify the quantities of wastes in the main structure of the Pile to assist in forward planning for decommissioning beyond the fuel removal stage.

7. Interactions with other CRP members

During the course of the CRP contact was made with the following CRP members for technical support in the areas below:

- J Dadoumont, SCK•CEN, Mol – decontamination techniques, dismantling techniques, use of VISIPLAN.
- S Mikheykin, Radon, Russian Federation – decontamination techniques
- G Rindahl, IFE, Halden Project – virtual reality

The purpose of the contacts was to gain the latest information on emerging technologies in D&D during the course of a desk-top review for application to UK decommissioning problems.

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