

Decommissioning techniques for research reactors

*Final report of a co-ordinated research project
1997–2001*



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FOREWORD

In its international role, the IAEA is faced with a wide variety of national situations and different availability of technical, human and financial resources. While it is recognised that nuclear decommissioning is a mature industry in some developed countries, and may soon become a routine activity, the situation is by no means so clear in other countries. In addition, transfer of technologies and know-how from developed to developing countries is not a spontaneous, straightforward process, and will take time and considerable effort. As mandated by its own statute and Member States' requests, the IAEA continues to respond to its Member States by monitoring technological progress, ensuring development of safer and more efficient strategies and fostering international information exchange.

Previous co-ordinated research projects (CRP) conducted respectively from 1984 to 1987, and from 1989 to 1993, investigated the overall domain of decommissioning. In those CRPs no distinction was made between decommissioning activities carried out at nuclear power plants, research reactors or nuclear fuel cycle facilities. With technological progress and experience gained, it became clear that decommissioning of research reactors had certain specific characteristics which needed a dedicated approach. In addition, a large number of research reactors reached a state of permanent shutdown in the 1990s and were candidates for prompt decommissioning. With the progressive ageing of research reactors, many more of these units will soon become redundant worldwide and require decommissioning. Within this context, a CRP on Decommissioning Techniques for Research Reactors was launched and conducted by the IAEA from 1997 to 2001 in order to prepare for eventual decommissioning.

Concluding reports that summarized the work undertaken under the aegis of the CRP were presented at the third and final Research Co-ordination Meeting held in Kendal, United Kingdom, 14–18 May 2001, and are collected in this technical publication. Operating experience in real-scale applications, lessons learned, key results in laboratory scale or pilot scale research, and validation of 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 co-operation and warm hospitality of the institutions that hosted the RCMs. Special thanks are due to P. Ernst, Canada, who 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

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1. INTRODUCTION

Research reactors play a significant role in the field of nuclear science and technology. Since early prototypes were designed and put into operation in the 1940s, the number of research reactors worldwide increased rapidly as the result of developments in the nuclear industry in general and nuclear power programmes in particular. Research reactors also contributed substantially in the area of non-power applications such as radioisotope production for applications in nuclear medicine, agriculture and industry; neutron beam research; training; neutron activation analysis, material development and neutron radiography. In total, more than 600 research reactors have been built and operated worldwide [1, 2].

The picture has changed considerably over the last 5–10 years with the reduced demand for many of the aforementioned programmes, maturity of the nuclear industry, increased competitiveness of the radioisotope market, increased competition for R&D funds, and escalating operation and maintenance costs of ageing reactors. The number of redundant reactors gradually increased to the point that the number of shutdown/decommissioned reactors is comparable now to that of operational ones [1–3]. The trend has been clearly visible for a number of years and there are no signs of a reverse of this trend. This inevitably means that more attention should be given to decontamination and dismantling of these older research reactors.

According to comprehensive statistics, over 650 research reactors have been built or are in the construction or planning phase throughout the world [2]. Of these reactors, over 350 have been shut down and decommissioned to various stages. Well over 200 research reactors operating today are already 30 years old and will become likely candidates for decommissioning in the near term. Many of these reactors are located in Member States where appropriate decommissioning experience may not readily be available. It should be noted that research reactors are ubiquitous, making their decommissioning a real international issue.

Within the overall overview of the nuclear industry, decommissioning plays a special role. It should be noted that, because of the natural life cycle of nuclear facilities, decommissioning was the last component of a nuclear programme that necessitated attention in most countries. Even today, there is a perception in certain environments that decommissioning is an easily manageable activity, which can be implemented at any time needing no advance planning. This inaccurate perception was established in the years of a flourishing nuclear industry, when the focus was on construction and operation of nuclear reactors, and planning for shutdown and decommissioning was relegated to a distant (yet unknown) future year.

Specific attention to the subject of research reactor decommissioning is considered necessary because of the unique aspects of research reactor facilities when compared to other nuclear facilities. Significant aspects of research reactors making decommissioning activities distinctly different from other nuclear facilities include:

- the broad spectrum of research reactor types including prototype reactors;
- the broad range and specificity of experimental work carried out in research reactors (e.g. impact of government policy, direction of national programmes); and
- the proximity of some research reactors to the public domain.

In particular, attention to and planning for eventual decommissioning of research reactors has been generally poor in most countries while reactors were being designed, constructed, operated and shut down. Plans for decommissioning were at best “a rough conceptual plan” in most countries and proper infrastructure was either missing or inadequate. This included the lack of decommissioning-oriented regulations, record keeping, waste management and disposal sites, expertise, training, and technologies. All of these aspects did not receive proper planning attention with the inaccurate perception that decommissioning could be accomplished quite readily with minimal planning and available resources. Complacency in decommissioning planning and implementation has resulted in undue delays, lack of funding availability and other resources and ultimately ended up with extra costs [2, 4].

Experience has shown that decommissioning can be undertaken without any deleterious effect on the safety of site personnel or the public, or any identifiable adverse impact on the environment, provided that decommissioning activities are undertaken in accordance with a properly formulated plan. The potential or actual radiological hazard associated with reactors may require the application of special techniques and procedures during decommissioning. Therefore it is essential that the identification and reduction of radiological hazards is given primary consideration in the preparation of the decommissioning plan and implementation of decommissioning activities. In particular, an item that is becoming of general concern for the Member States is the need to resolve the problem of disposing of radioactive waste arising from decommissioning of such reactors. Lack of radioactive waste disposal facilities remains an impediment to achieving a decommissioning end state for many nuclear facilities. Also, the costs of complete dismantling of a relatively large research reactor are significant for a small country, particularly if decommissioning funds were not established and secured in advance of final shutdown.

Another important consequence of poor planning in decommissioning of research reactors was the tendency to approach each decommissioning project as a first-of-the-kind project, even when experience was already available elsewhere. With time, it became clear that experience was being gathered not only from active decommissioning projects but also from major decontamination and refurbishment of operating reactors. This is in turn reflected in early and more accurate planning for decommissioning as highlighted by national papers presented in this TECDOC.

The IAEA has for years provided practical and regulatory guidance on decommissioning with the objective of fostering exchange of information and know-how and harmonising approaches and strategies. To this end, mechanisms such as dissemination of documents and reports [2, 4–8], training courses or direct assistance to Member States [9, 10] have been applied. Another useful mechanism is the Co-ordinated Research Project (CRP), which is relevant to the work presented in this document, and will be described in the following sections.

2. CO-ORDINATED 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 imperative 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 globalization 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.

This CRP on Decommissioning Techniques for Research Reactors is the ideal continuation of two CRPs conducted, earlier in 1984–1987 and 1989–1993 in the field of decontamination and decommissioning of nuclear facilities. The main results of these CRPs were collected in TECDOCs for free distribution to Member States [11, 12]. It should be noted however that the two CRPs were for nuclear facilities in general and not specific to a particular type of facility. For the reasons discussed in Section 1, the decommissioning of research reactors required special attention. It is believed that this information gap was closed as the result of the CRP which is being described in this document. Also, as decommissioning covers a broad, multi-disciplinary field, it is felt now that, to be cost effective, a CRP should address specific technical disciplines (like the CRP on New Methods and Techniques for Optimization of Decontamination for Maintenance or Decommissioning [13]) and/or specific types of nuclear installations (such as research reactors in this case).

3. SCIENTIFIC SCOPE AND PROJECT GOALS

The objective of this CRP was to promote the exchange of information on the practical experience gained by Member States in decommissioning or operation, maintenance, and refurbishment activities which would be eventually related to the decommissioning of research reactors. Special emphasis was given to the development/adaptation of methods and approaches for optimization of the decommissioning process. The scope of the project included several technical areas of decommissioning rather than focusing on a single aspect of it. It was felt that this format would generate more awareness of the integrated approach to decommissioning. In particular, the scope included the following:

- design, construction and operational features to assist in final decommissioning;
- planning for decommissioning, including technical solution assessment;
- decommissioning strategies and their technological implications;
- radiological and physical characterization;
- dismantling technology;
- decontamination technology;
- remotely operated equipment;
- means to reduce occupational exposures;
- waste generation and management, including clearance of solid materials;
- restricted and unrestricted site release, including final surveys;
- costs and financial provisions;
- safe enclosure of shutdown reactors, including long-term integrity of buildings and systems;
- decommissioning experience; and
- ageing management and refurbishment experience.

Ten Research Agreements and four Research Contracts stipulated with institutions from thirteen different Member States were included in this CRP. Under the aegis of this CRP, three Research Co-ordination Meetings were held, respectively, in Mumbai (India, 16–20 February 1998), Taejon (Korea, 14–18 February 2000) and Kendal (United Kingdom, 14–18 May 2001). This TECDOC includes final project reports presented at the third and final meeting.

4. SUMMARY OF MAJOR TECHNICAL ACHIEVEMENTS

As said above, this CRP investigated practically the whole range of decommissioning-related activities and technologies for research reactors. The following highlights specific R&D areas covered by the CRP and pertinent individual projects.

Planning 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. Aspects relevant to and facilitating (or hindering) decommissioning were investigated in the course of the PARR-1 refurbishment project, Pakistan. The design of a new spent fuel storage facility was described in an Egyptian project and took into account features to facilitate future decommissioning. Planning activities were also described in a number of specific decommissioning projects.

Radiological characterization was extensively performed at and reported by: CIRUS reactor, India with the purpose of generating a database for future decommissioning; AM, Russian Federation, with the purpose of collecting preliminary data for a decommissioning project and a focus on activated graphite and reactor vessel; Nuclear Power Demonstration (NPD), and Whiteshell Laboratories, Canada; DR-2, Denmark.

Safety assessments of the radiological inventory as the basis for the selection of decommissioning strategies were conducted at and reported by: DR-2, Denmark; NPD and Whiteshell Laboratories, Canada (a rare case of in-situ disposal strategy); the ICI reactor in UK; the Indonesian TRIGA and the two Korean reactors, TRIGA Mark-II and TRIGA Mark-III.

Decontamination technologies were investigated in a number of projects namely: the IRT-M reactor in Belarus (the very scope of the project); the AM reactor in the Russian Federation (with a focus on contaminated graphite and primary circuit components); CIRUS, India; and the Indonesian reactor (with a focus on the Aluminium graphite reflector surface and the stainless steel primary coolant heat exchanger).

Dismantling technologies were addressed in a number of projects and in particular: JEN-1, Spain (thermal and mechanical underwater metal cutting with remotely operated equipment); and BR-3, Belgium (remote mechanical cutting of the reactor pressure vessel). These two projects exhibited common issues such as poor water visibility.

The projects underway at Argonne National Laboratory (ANL, USA) served for the development of numerous innovative decommissioning technologies addressing radiological characterization, waste management, cutting, decontamination etc. Further deployments are needed to determine which technologies are ready for use as baseline technologies over existing ones.

From another perspective this CRP addressed a variety of research reactor types. Pool-type reactors include the 3-MW(th) JEN-1 (Spain), the IRT-M reactor in Belarus and the 5-MW(th) PARR-1 (Pakistan). Various models of TRIGAs are addressed by decommissioning projects i.e. the 250-KW ICI Mk 1 (UK), the 1000-KW (being upgraded to 2000-KW) Mk 2 in Bandung (Indonesia), and the two Korean reactors (a 250-KW TRIGA Mark-II, and a 2-MW TRIGA Mark-III). Tank-type, light water reactors include ETTR-1 (5-MW(th)) of the Soviet design WWR, Egypt), DR-2 (5-MW(th), Denmark), UTR 300 (the 300-KW(th)) Argonaut reactor in Scotland) and BR-3 (the prototype 40-MW(th) PWR in Mol, Belgium). Heavy water reactors include NPD (a prototype CANDU in Canada), CP-5 at Argonne National Laboratory, USA, and CIRUS at Bhabha Atomic Research Centre, India. AM in Obninsk, Russian Federation, is a reactor with unique features including large amounts of graphite.

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 (BR-3, Belgium; ICI TRIGA and Scottish Universities Research Reactor, UK; CP-5, USA; JEN-1, Spain; IRT-M, Belarus). Others decommissioning projects were at the planning stage (AM, Russian Federation; NPD, Canada; Korean TRIGAs; ETTR-1, Egypt; and DR-2, Denmark). Lastly, some other projects where existing operating reactors were refurbished for life extension drew on lessons learned for future decommissioning purposes (PARR-1, Pakistan; TRIGA reactor, Indonesia; and CIRUS, India).

5. PENDING ISSUES

As one result of the CRP, a few critical issues — generally common to several decommissioning projects addressed by the CRP — became apparent and are discussed in the following paragraphs.

Until recently, decommissioning of research reactors was not given adequate priority by operators and other decision makers. Particularly for facilities owned by public institutions it was generally assumed that the State would take over after permanent shutdown of a nuclear plant and resolve any “open” decommissioning issues. Also, it was generally believed that resources required for the decommissioning would be readily available when needed. The immediate result of this approach was that no planning or inadequate planning for decommissioning was performed including inadequate infrastructures (e.g. no decommissioning-oriented regulations), lack of funding, poor experience /expertise in decontamination and dismantling technologies, absence of specialised contractors and/or a general lack of interest. It should also be mentioned that political changes e.g. the creation of new independent states led to disruption of traditional links including the availability of services and specialists. Until a few years ago, studies on decommissioning of certain types of nuclear facilities were rare. Luckily, this situation is now quickly changing, mostly as the result of international co-operation in which the IAEA plays an essential role. For example, in recent years a number of Member States have established mechanisms to collect and earmark funds for decommissioning during plant operation. It is unfortunate that some Member States are lagging in this planning due to inter alia conflicting needs and priorities, and limited resources. Funding for decommissioning has proven particularly difficult for facilities owned by public institutions (e.g. in Indonesia and Russian Federation).

Another area where more progress is essential — so denoting a serious unsolved issue — is the cultural change required by decommissioning. Inadequacies in this area may be even more serious than the lack of adequate technologies (regardless of financial limitations, the latter can be found at the end of the day in the international market). Related issues typically include poor co-ordination among parties, over-centralization, unclear and complicated regulations and administrative procedures, and ultimately domestic conflicts and lack of national consensus. These conditions affect timeliness, efficiency (e.g. costs) and ultimately safety of decommissioning activities. Another related problem — not uncommon even in industrialised countries — is personnel qualifications and motivation. Sometimes it is difficult or even impossible for a team of operators previously responsible for the operation of a research reactor to adjust to the reality of a dismantling project. The unfortunate common situation of research reactors remaining in a shutdown state for many years with a lack of prospects (continued operation, refurbishment and/or decommissioning strategies) is not conducive to well planned decommissioning projects. Loss of records or of staff's historical memory are critical factors in this regard.

There should be serious recognition that operational and research staff knowledge will be beneficial during decommissioning, particularly during the transition from operation to decommissioning. No-action or undue delays in this transition are a serious concern particularly due to the ageing of operational staff in many research reactors. Mechanisms should be in place to expedite transition including the final establishment of a comprehensive set of decommissioning records. Nevertheless, to plan for decommissioning and the final end state it is important that a significant body of professional decommissioning expertise be developed and maintained. Such decommissioning expertise is specialised and certainly different from the experience base required during the operating period. Training or re-training of decommissioning operators is another important consideration, particularly in deferred decommissioning.

Organization and management of decommissioning projects is another area requiring attention. Hindrances to smooth progress of decommissioning may include poor record keeping and lack of information on technologies/experience available in a given country. These aspects may require considerable organizational efforts prior to commencement of decommissioning. Also, there is often a poor understanding that decommissioning unlike R&D has a beginning and an end. The perception that the objective is “to work yourself out of a job” is to be dealt with carefully in a well planned decommissioning project.

Despite the maturity of the nuclear decommissioning technology/industry at least in industrialised countries, there are still a few areas where progress is hindered and more work is needed. A few such areas are discussed in the following.

One is the achievement of international consensus on clearance levels. This is a field where the IAEA is actively involved together with other international agencies. The promulgation of internationally accepted clearance levels is not only a necessity for national decommissioning projects, but also a strong requirement in order not to have undesirable consequences associated with the release and/or reuse of decommissioning waste/materials. There have been already cases where materials released as “non-radioactive” in one country have been classified as “radioactive” in another country and sent back. The potential for significant disruption in world trade is clear to everybody. However, in Member States having limited decommissioning programmes (e.g. Indonesia) the likely destination of small amounts

of decommissioning materials/wastes will be storage, restricted use or research. Currently, case-by-case clearance criteria seem prevailing in many countries.

Another area where technological developments are needed is the treatment and disposal of decommissioning wastes, particularly medium and high activity materials. This aspect is of course part of the more general issue of treatment and disposal of any radioactive wastes, but the specific features of decommissioning wastes provide additional urgency to the solution of this issue. Member States not having established technologies for medium and high level waste management will have to opt for interim storage of those wastes. Special materials e.g. beryllium need attention (see DR-2 project, Denmark) in that there are up to now no clear disposal routes for those materials. Management of graphite wastes has unique features due to their radioisotopes having very long half-lives. As investigated at the AM project, one possibility is to wait for decay of short lived radioisotopes (~100–150 years) and then place radioactive graphite into dry subsurface repository for thousands years ‘ storage. Another investigated alternative is the burning of activated, non-contaminated parts of graphite. It was estimated by those responsible for the AM project that this approach — particularly for the low amounts of graphite resulting from the decommissioning of research reactors — will lead to negligible impact on the environment. Radiation swelling of graphite column was mentioned as a serious complication for the IRT-M project. Generation of large amounts of wastewater in soil washing and unexpected precipitation in an electrokinetic method were critical issues in the TRIGA soil remediation project in Korea.

Despite the general maturity of the nuclear industry, there are still technological improvements to reach in other areas. Lessons learned from the JEN-1 project include operational problems with a melting facility (adaptation to higher temperatures, poor decontamination factors) and issues in thermal processes (need for aerosol control, poor visibility, application to thick items). Lack of radionuclide inventory detail, in particular in reactor areas/components, is another issue potentially hindering progress (see NPD project, Canada).

Experience of decommissioning nuclear facilities to date, as well as related studies, has shown that consideration should be given to decommissioning during the plant design and construction phases. Numerous reports and studies have highlighted the costs and complexities involved in decommissioning existing nuclear facilities. The objectives of including design features to facilitate decommissioning are to reduce occupational exposures, minimize waste generation and other environmental impacts, simplify dismantling procedures and reduce costs. A variety of concepts have been used or proposed for inclusion at the design stage to facilitate various decommissioning activities including decontamination, dismantling, and removal or segmentation of components. It is highly desirable that designers of new plants be aware of the issues, strategies and techniques involved in decommissioning. Lessons learned from decommissioning projects are invaluable for creating an awareness of “decommissioning-friendly” features for future designs.

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 developing project and country specific technologies (supposedly at the lowest cost), and purchasing technologies in the open market remains a serious challenge for many countries. Timely allocation of decommissioning funds is important to alleviate these concerns and minimize delays in project implementation.

6. PROJECT OUTCOME

It is felt that the IAEA project 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 need for timely planning for and implementation of decommissioning. In some Member States there are research reactors 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, entails deterioration of structures and components, and may in the longer term have very serious safety implications. The IAEA project is expected to offer the Member States the opportunity of considering financial and other impacts of decommissioning research reactors, 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.

In more general terms, the project will contribute to enhancing Member States' overall organizational capabilities. As decommissioning is a multi-disciplinary process, the project will stimulate Member States to develop an integrated approach to decommissioning by making use of resources available both domestically and internationally. In this regard, the project impact may go far beyond the scope of research reactor 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 to be of an international nature in the near future. 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. CRPs are the typical mechanisms for implementing such a strategy. Co-operation 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.

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¹ References include selected IAEA publications on decommissioning of research reactors or papers describing the IAEA programme in this field.

Decontamination of Belarus research reactor installation by strippable coatings

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Abstract. The goal of this study was to develop new strippable coatings using water-based solutions of polyvinyl alcohol and active additives for decontamination of research reactor equipment. The employment of strippable coatings makes it possible to minimize the quantity of liquid radioactive waste. The selection of strippable decontaminating coatings was carried out on the basis of general requirements to decontaminating solutions: successfully dissolve corrosion deposits; ensure the desorption of radionuclides from the surfaces and the absence of resorption; introduce minimal corrosion effect of construction materials; to be relatively cheap and available in reagents. The decontaminating ability and adhesion properties of these coatings depending on metal and deposit sorts were investigated. Research on the chemical stability of solid wastes was carried out. The data obtained were the base for recommendations on waste management procedure for used films and pastes. A full-scale case-study analysis was performed for comparing strippable coatings with decontaminating solutions.

1. Introduction

1.1. Research reactor of Belarus Academy of Sciences

The nuclear reactor of IRT-type was operated for 25 years at the former Institute of Nuclear Power Engineering Academy of Sciences of Byelorussian SSR. The reactor with a design power of 2000 kW was put into operation in April 1962. The reactor was operated at that power level up to 1971 using rod-type fuel elements with uranium enriched to 10% in ^{235}U as fuel and a graphite reflector.

In 1971 the reactor was modernized to increase its power to 4000 kW and improve its design while preserving the main designed principles peculiar to a pool-type reactor. During the modernisation the fuel assemblies (FA) were changed to IRT-2M type containing uranium enrichment to 90 % in ^{235}U reflected by beryllium.

During the operating period from 1962 to 1987 the total energy output of the reactor was 152.8 GW·h during a total operating time $68.7 \cdot 10^3$ hours including 19 GW·h and $15.5 \cdot 10^3$ hours before its modification. Power and time regimes of the reactor's operation were defined by the experimental programs mainly by the loop experiments in the core of the reactor.

The design of the core included the possibility of installing special experimental loops in place of 4 FA in the core of the reactor for carrying out investigations. The design of the core permitted up to 8 horizontal experimental channels to be installed.

The reactor has been used for carrying out investigations in the field of nuclear and radiation physics, neutron activation analysis, radiation chemistry and radiobiology. From the beginning of the 70's radiation construction material applied to the problem of using dissociating gases as a coolant in NPP was defined as the main trend of scientific research. Capsule and loop installations were created at the reactor for studying properties of nitrogen tetraoxide (N_2O_4).

During operation, the fuel loading in the core was 28-33 FA. The average burnup of ^{235}U in the fuel was 40%, maximum burnup was 53% in 10% of the spent FA, 56 FA were completely used and 29 FA were partially used.

In 1987 the reactor was shut down for inspection of its technical condition, developing and implementing measures to ensure its further safe operation and evaluation of proper physical parameters of the reactor to the demands of the research work. The evaluation of the technical condition of the reactor as well as the potential of its further use showed that the main reactor components (vessel, experimental channels, heat exchange facilities), and the components of a number of systems had surpassed the end of their physical service-life or were at the limit of their use, had difficulty correcting defects and therefore were in need of replacement.

For these reasons the reactor ceased operation in 1989, when the decommissioning project for the reactor was developed. The project envisaged dismantling, decontamination or disposal of the components and the systems of the reactor which were contaminated, excluding the concrete mass of the biological shield for the vessel, with the remainder of the main reactor building turned into office space [1].

1.2. Project objectives

Decontamination operations of nuclear facilities necessarily result in the generation of airborne, liquid and solid radioactive waste. Obviously, it is desirable to use processes that produce the smallest volumes of additional waste and wastes that are the most amenable to volume reduction and conditioning.

A potentially attractive decontamination method that reduces liquid radioactive waste volumes is to use strippable coatings. They are sprayed or painted on to surfaces and allowed to dry. The solvent then evaporates to leave a thin plastic film or layer, which is of sufficient strength to remain intact when peeled or stripped off.

The method offers the advantage that contaminated surfaces can be decontaminated *in situ*, with contaminated waste only occurring in compact and solid form. There is no washing or flushing of decontaminated surfaces and hence there is no dilution or further propagation of activity or contamination.

This report describes experiments aimed at demonstrating the applicability and effectiveness of strippable coatings as a means of removing contamination from the surface of equipment.

The programme of work included:

- development of optimized compositions of strippable coating components (matrix, aggressive additives, decontamination solutions);
- investigations of mechanical, adsorption and decontamination properties of strippable coatings;
- development of strippable coatings which included protective and corrosion inhibiting characteristics;
- investigations of chemical stability of used coatings and development of waste management methods;
- modifying of compositions intended for oxide deposit removal;

- adaptation of standard degreasing and cleaning compositions for decontamination purposes;
- a full scale study to compare strippable coatings with more traditional chemical decontamination techniques.

2. Results and discussion

2.1. The development of strippable films

Investigations into strippable coating compositions started with preliminary tests with the objective of selecting a suitable matrix material which met the following requirements:

- water is preferred as the solvent for the matrix;
- it has to be resistant to mineral acids and alkalies;
- it has to be compatible with organic acids;
- it has to be miscible with emulsifiers, softeners and thickeners.

Such film formers as polyvinyl alcohol (PVA) and polyvinyl acetate were investigated. The polyvinyl alcohol is soluble in water, but requires some alcohol for soaking or softening. The polyvinyl alcohol is decomposed by biological methods over a long period of time. Application of polyvinyl alcohol in wet conditions is limited because of poor drying of the foil.

Water-soluble organic compounds with hydroxyl groups, such as glycerol, ethylene glycol, and polyethylene glycol were added as plasticizers to the solution of polyvinyl alcohol.

The water dispersion of polyvinyl acetate with polymer content about 50–55% and particles of 0.052.0 μm in size was diluted with water and acetone.

Such compounds as H_3PO_4 , H_2SO_4 , HNO_3 , KOH , NaOH , organic acids and their salts, oxidants and EDTA. 1-hydroxyethylendiphosphonic acid and/or its salts (HEDTA) were investigated as pickling or decontaminating agents.

The advantage of organic acids is the absence of corrosion influence on metallic surfaces. The joint action of an organic acid and complexing agent possesses increased decontaminating ability in comparison with the separate actions of these components. There are several explanations of this phenomenon. It is believed that the organic acid reacts with cations of the corrosion deposit and transfers them into solution. Then the complexing agent reacts with the cations by forming a complex compound and releasing anions of acid for the further reactions.

The actual complexing ability of a composition is higher than the theoretical abilities of the composition components. Obviously it is explained by the formation of complicated complexes with the metal ions. The complexes submit to other stoichiometric proportions, which cannot be determined by the additivity rule. As a result, the consumption of the reagents is reduced.

The experiments to develop strippable coatings showed that the addition of decontaminating and pickling agents to the film-forming composition resulted often in destruction or coagulation of polymer compositions. Most frequently the phenomenon was observed when using polyvinyl acetate, which has a low degree of compatibility with the

additives investigated. At the same time polyvinyl alcohol based films showed high compatibility with these chemicals without the film quality being appreciably impaired.

2.1.1. Investigation of adhesion properties of the films

From a large number of test mixtures with different concentrations of additives, the choice finally fell on compositions, which proved to be optimum with respect to:

- (a) foil formation;
- (b) elasticity;
- (c) tearing strength;
- (d) adhesion to the background;
- (e) drying time.

The chemicals included in these mixtures proved, in general, to be suitable for degreasing and pickling metallic materials.

The foil mixtures were applied to sample sheets made of stainless steel (X18H10T), carbon steel, aluminium (AlMg₃), brass and brick by brushing and spraying.

The definition of adhesion of decontaminating film-forming compositions to samples was performed by the standard method of parallel incision [2]. The results of the tests were in values of adhesion in points according to a 4-point scale. The highest point corresponds to the least adhesion of coating to the background. The results of the tests are listed in Table I.

The results achieved showed that developed film-forming coating possessed reduced adhesion to metal surfaces but adhesion to the porous surface of brick was too high.

Table I. The adhesion of strippable films

DECONTAMINATING COMPOSITION	STAINLESS STEEL	CARBON STEEL	ALUMINIUM	BRASS	BRICK
H ₂ SO ₄ , EDTA, polyvinyl alcohol, glycerol	2	3	3	2	1
H ₃ PO ₄ , tartaric acid, C ₂ H ₅ OH, polyvinyl alcohol	3	3	3	3	1
H ₃ PO ₄ , HEDPA, C ₂ H ₅ OH, glycerol, polyvinyl alcohol	3	2	2	2	1
KOH, carboxymethyl-cellulose, glycerol, polyvinyl alcohol	3	3	4	4	2
NH ₄ F, H ₃ PO ₄ , HEDPA, C ₂ H ₅ OH, polyvinyl alcohol	2	3	3	3	2
EDTA, triethanolamine, NaOH, polyvinyl acetate	3	2	3	3	1
NaOH, EDTA, polyvinyl alcohol	4	4	4	4	2

2.1.2. Testing of strippable foil on painted metal

During dismantling operations of the IRT-M research reactor and accompanying measurements it was shown that the radionuclides were incorporated into paint layers. Preliminary laboratory tests on the removal of paint from the surface of carbon steel were performed with the objective of selecting optimal strippable film-forming composition for decontamination of painted equipment.

The carbon steel plates were painted and then allowed to dry for 3 months. Then the samples were brushed by film-forming compositions. Hardened coatings were removed after 20 hours of drying. The quantity of removed colour was determined by a gravimetric method. The cycle of treatment was repeated once again. The results obtained are presented in Table II.

The data showed that alkaline compositions, especially KOH, removed the colour satisfactorily. The compositions could be used for removing grease deposits from metal surfaces.

Table II. Removing of oil colour

DECONTAMINATING COMPOSITION	REMOVED COLOUR. %		
	1 ST CYCLE	2 ND CYCLE	2 CYCLES
12% PVA + 5 % NaOH	38.6	21.9	60.5
	26.7	42.5	69.2
	19.4	39.3	58.7
12% PVA + 10 % NaOH	41.8	38.1	79.9
	46.8	42.9	89.7
	63.5	31.6	95.1
12% PVA + 5 % KOH	30.1	44.1	74.2
	22.3	34.9	57.2
	54.0	22.3	76.3
12% PVA + 10 % KOH	64.9	26.9	91.8
	79.1	17.6	96.7
	81.9	16.8	98.7

2.1.3. Decontamination of metal surfaces

Before starting decontamination of real equipment different decontamination strippable films were tested on the simulated samples of different construction materials. The samples of stainless steel, carbon steel, and carbon steel with grease and rusted deposits were selected in order to determine the decontamination effect of the strippable decontamination coatings.

The data showed that the presence of oxidised layers and grease deposits on steel surfaces decreased the decontaminating effectiveness of strippable films significantly (4–7 times). The decontamination of rusted surfaces is the most difficult.

The results of the tests indicated that a compound consisting of 1-hydroxy ethyldendrophosphonic acid and/or its salts was the most effective decontamination agent.

2.2. Development of decontaminating pastes

The investigation of decontamination ability of pastes was carried out with the objective of optimising decontaminating compositions and technologies. Decontaminating pastes combine the advantages of strippable films and leaching-desorbing pastes. The decontaminating pastes can be deposited on the surfaces of components as viscous substances (for instance plaster solutions) and can be removed as a dry hard mass.

By using the new films in the decontamination of the ventilation systems components, a decontamination factor of 10–40 was obtained for one treatment cycle depending on the form of the contamination. The results achieved are shown in table IV.

Table III. Decontamination of steels

Decontaminating strippable film (%)	Decontamination factor (DF)			
	Stainless steel	Carbon steel	Carbon steel with greased deposits	Carbon steel with rusted deposits
HEDPA - 1.5 Na ₃ PO ₄ - 2 glycerol - 8 PVA - 12.5	20.0	6.8	2.5	1.3
sulphamic acid - 0.5 propyleneglycol -10 (NH ₄) ₂ SO ₄ - 1 PVA - 10	24.6	8.8	3.7	1.8
NH ₄ F - 1 H ₃ PO ₄ - 1.8 HEDPA - 1.5 glycerol - 10 PVA - 12	26.6	24.4	21.1	4.9
HEDPA - 2 H ₂ C ₂ O ₄ - 2.5 NH ₄ F - 0.5 glycerol - 8 PVA - 11	30.3	16.8	7.6	4.5
NaOH - 10 propyleneglycol -10 sulphamic acid - 1 PVA - 12	19.8	5.3	4.5	1.3

Table IV. Decontamination of ventilation systems using special films

Equipment. material	Counts before decontamination (cm ⁻² ·min ⁻¹)	Counts after decontamination (cm ⁻² ·min ⁻¹)	DF
motors (painted cast-iron)	400	62	6.5
	904	48	18.8
	350	30	11.7
	310	30	10.3
	480	66	7.3
	950	82	11.6
	330	54	6.1
	1800	60	30.0
air channels (galvanized steel)	2700	85	32.0
	2000	50	40.0
	120	7	17.1
	97	5	19.4
	90	4	22.5
working wheels (painted steel)	500	55	10.0
	440	39	10.3
	302	40	7.6

Decontaminating pastes are heavily concentrated dispersal systems possessing structural properties. The application of decontaminating pastes reduces the consumption of reagents and the volume of liquid radioactive wastes. Decontaminating pastes should comply with requirements of decontaminating solutions.

2.2.1. Absorption properties of lignin and clinoptilolite

The high content of the absorbents in decontaminating pastes contribute to firm immobilisation of radionuclides in solid wastes. The absorption abilities of the clinoptilolite and cation-exchange resin KU-2 (Dowex-50) are well known. Tests were performed to evaluate the absorption of ^{137}Cs and ^{90}Sr radionuclides by lignin.

The absorption experiments were batch contacts with agitation of the solution and absorbent. The extent of absorption was determined by the change in concentration of the radionuclide in solution and calculated in terms of the distribution coefficient K_d .

The amount of ^{90}Sr in solution was determined by radiochemical separation using a carrier followed by determination of the activity of the separated precipitate and yield of the carrier.

The experimental results are presented in Table V. As the presented data show, an equilibrium stage in the lignin-solution system could be reached in 6–7 hours. This time interval corresponds to the hardening time of the majority of film-forming compositions, and significant absorption of radionuclides, especially ^{90}Sr , takes place during the process.

Table V. Kinetics of ^{137}Cs and ^{90}Sr absorption by lignin

TIME HOURS	K_d FOR ^{137}CS			K_d FOR ^{90}SR		
	pH 3.0	pH 7.0	pH 9.0	pH 3.0	pH 7.0	pH 9.0
1	0.3	0.5	0.5	1.8	5.3	6.8
2	0.6	0.8	0.7	1.8	6.1	10.7
3	0.9	2.8	1.4	2.6	7.7	11.4
4	1.8	4.3	2.5	3.3	8.1	16.4
5	3.9	6.9	7.2	4.6	8.6	28.3
6	7.8	7.8	8.6	7.3	10.6	28.3
7	7.8	8.3	10.6	8.2	12.0	29.4

2.2.2. Investigation of adhesion properties of the pastes

The adhesion of pastes to the surfaces is less than the adhesion of films to the surfaces. This effect is achieved by the addition of different fillers such as lignin, clinoptilolite, or ion-exchange resins into the film-forming composition.

Lignin is a natural polymer that is contained in timber (~ 30 %). It can be emitted during the process of wood polysaccharides hydrolysis. The wastes of the hydrolysis industry (hydrolysis lignin) and paper production (lignosulphonic acids) are changed considerably and they are hard to use. In biomass lignin is generally second in quantity after cellulose. The lignin molecule is a large and complex polymer, which is very efficient in forming chelates with cations, even with monovalent ions like caesium.

Besides, the powders of lignin and absorbents could be used as fillers. The tests were performed to determine the adhesion of pastes with different amounts of fillers to stainless steel. The film-forming composition containing H_3PO_4 , tartaric acid, HEDPA, glycerol, ethanol and polyvinyl alcohol was used as a basic matrix for the tests. Lignin, clinoptilolite and cation-exchange resin KU-2 (Dowex-50) were used as fillers. The results are presented in Table VI.

The data showed that the addition of more than 25 % of the absorbents decreased the adhesion of the pastes to the stainless steel.

Table VI. Adhesion of decontaminating pastes

CONTENT OF ABSORBENT (%)			ADHESION POINTS		
CLINOPTILOLITE	KU-2	LIGNIN	CLINOPTILOLITE	KU-2	LIGNIN
0	2.9	5.9	2	2	2
7.4	4.7	7.2	3	3	3
9.9	10.8	19.1	3	4	3
25.4	12.5	26.7	4	4	4
35.6	25.6	34.8	exfoliation		
43.7	31.6	46.2	exfoliation		

2.2.3. Decontaminating capability of the pastes

Investigations of different paste composition efficiencies were carried out to optimize decontamination technology. The decontaminating capability of the pastes containing lignin and clinoptilolite was tested on steel (carbon and stainless) and aluminium surfaces. Decontaminating pastes contained aggressive agents such as alkaline (composition 1) and acids (compositions 2–4), and also complexing and stabilising additives. The results are presented in Table VII and Table VIII.

As the data show, the addition of lignin did not intensify the decontaminating capability of the pastes. At the same time the decrease of the paste adhesion to metallic surfaces was observed when lignin content was increased. Paste of composition 4 possesses the least adhesion up to the exfoliation. This reduces labour expenses during the removal of hardened decontaminating pastes.

It should be noted that in all cases of decontamination the sample surfaces became clear and smooth, without corrosion deposits.

It is evident that the creation of a strippable coating with a universal composition is difficult. A series of different compositions is necessary for successful decontamination.

In the experiment parts of rusty painted tools were decontaminated. The first decontamination cycle was carried out to remove the rust and some of the contamination, including corrosion deposits. The paste composition was: oxalic acid; 1-hydroxyethylidenediphosphonic acid; ammonium thiocarbamate; polyvinyl alcohol; lignin: $DF=2.1 + 0.8$; Consumption = 0.6 kg/m^2 .

Table VII. Decontaminating capability of the lignin pastes

COMPOSITION - (%)		CONTENT OF SORBENT (%)	MATERIAL	DF
	NaOH – 10.0 (NH ₄) ₂ HEDPA-2.0 carboxymethylcellulose –0.5 PVA – 11.5	2.7	Carbon steel Stainless steel	17.1 13.6
		6.8	Carbon steel Stainless steel Aluminium	17.6 12.6 29.2
2	H ₂ SO ₄ – 9.2 H ₃ PO ₄ – 8.4 NH ₄ F – 3.0 PVA – 12.0	8.6	Carbon steel Stainless steel	37.8 38.1
		21.0	Carbon steel Stainless steel Aluminium	12.2 22.4 16.8
3	H ₃ PO ₄ - 10.0; HEDPA – 1.0; tartaric acid– 3.0; PVA – 12.0	4.7	Carbon steel Stainless steel	27.9 30.8
		9.2	Carbon steel Stainless steel Aluminium	25.0 29.2 26.6
4	H ₂ SO ₄ – 9.2 H ₃ PO ₄ – 8.4 tartaric acid– 3.0 PVA – 12.0	5.8	Carbon steel Stainless steel	7.8 21.5
		12.3	Carbon steel Stainless steel Aluminium	8.9 23.8 13.5

The second cycle was aimed at the removal of the paint and the contamination attached to the paint. The composition of the decontamination paste was the same as used for the test reported in Table I: NaOH. EDTA-Na. Lignin; polyvinyl alcohol; DF=4.4+1.4

As the data show, new strippable compositions with a polyvinyl alcohol base and active additives are sufficiently effective for the decontamination of research reactor components.

2.3. Developing strippable coatings including protective and corrosion inhibiting characteristics

Protective films and coatings are used for the prevention of re-contamination of cleaned surfaces; the spreading of radioactive contamination onto other objects and to increase the decontamination efficiency.

Polymer films containing corrosion inhibitors are one type of protective coating. The presupposition for these films application for anticorrosion protection is their ability to provide simultaneously both a barrier and inhibiting protection of metal surfaces, especially after decontamination. The application of inhibiting protective films during decontamination results in a double protective effect: protecting the decontaminated surface from recontamination; and providing corrosion protection of the cleaned metal surface.

Table VIII. Decontaminating capability of the clinoptilolite pastes

COMPOSITION. (%)		ABSORBENT CONTENT (%)	MATERIAL	DF
1.	NaOH – 10.0 (NH ₄) ₂ HEDPA-2.0 carboxymethylcellulose –0.5 PVA – 11.5	0	Carbon steel Stainless steel	30.9 44.9
		7	Carbon steel Stainless steel Aluminium	31.3 56.8 35.8
2.	H ₂ SO ₄ – 9.2 H ₃ PO ₄ – 8.4 NH ₄ F – 3.0 PVA – 12.0	2	Carbon steel Stainless steel	10.8 24.6
		.6	Carbon steel Stainless steel Aluminium	14.9 30.9 19.8
3.	H ₃ PO ₄ - 10.0; HEDPA – 1.0; tartaric acid– 3.0; PVA – 12.0	3	Carbon steel Stainless steel	34.6 40.6
		.1	Carbon steel Stainless steel	33.8 48.2
4.	H ₂ SO ₄ – 9.2 H ₃ PO ₄ – 8.4 tartaric acid– 3.0 PVA – 12.0	9	Carbon steel Stainless steel	15.4 28.6
		.3	Carbon steel Stainless steel Aluminium	18.7 42.4 21.2

The type of contamination and resource availability influences the choice of film components.

The development of corrosion inhibiting films was carried out with the basic polymer composition containing polyvinyl alcohol solution, ethanol and glycerol. Inhibiting properties were provided by the addition of cheap chemical compositions such as phosphate, chromate, nitrate, triethanolamine and thiourea. The metal samples covered by the inhibiting coatings were placed in an aggressive medium for 4 months. The results indicate that phosphate and thiourea possess the most inhibiting properties.

2.4. Working out of combined decontaminating solutions

Investigations of the efficiency of different compositions were continued to optimize decontamination technology. The standard compositions for cleaning and degreasing of metallic surfaces of industrial equipment were also tested since they have to respond to the same demands as those used for decontamination. Two approaches were taken:

- (1) modifying of the decontaminating compositions used for oxide deposit removal to enhance their detergent effect by the addition of SF-2U preparation;
- (2) adaptation of the standard degreasing and cleaning compositions for the decontamination goals.

The compositions and their decontaminating ability are presented in Table IX.

As the presented data indicate, the acidic decontaminating compositions, which were modified by the addition of SF-2U preparation, possess good decontamination and detergent properties. The standard industrial cleaning and detergent solutions possess less decontaminating ability. But they have the doubtless advantage of a smaller corrosion impact on the metallic surfaces. Besides, it is possible to employ the latter compositions in combination with hydro-jetting devices for surface cleaning. It should be noted, however, that the alkaline composition (19) had a large corrosion impact on aluminium surfaces.

The standard industrial cleaning composition (10) and especially the modified decontaminating solution (12) provided good cleaning and decontamination of aluminium surfaces, removing both mud-greasy and radionuclide deposits. As a result, the aluminium surfaces were clean, bright, and without any marks of chemical corrosion.

2.4.1. Investigations of detergent effect of solutions

Thus, the results presented in Table VIII are the sum of the two processes: decontamination and degreasing. Separate estimations of the corrosion impact and detergent effect of the solutions were carried out, especially in the case of aluminium where the removal of the deposits was accompanied by the removal of the upper layer of the base metal.

The detergent effect of the compositions was estimated by the formula:

$$DE = (P_1 - P_2) / (P_1 - P_0) \times 100\%$$

where:

P_0 is the weight of clean sample (g).

P_1 is the weight of dirty sample (g).

P_2 is the weight of a sample after washing (g).

The data show that the tested solutions (enumerated in Table IX) provide a high detergent effect (85–97%).

2.4.2. Investigation of the corrosion impact of decontamination solutions

Corrosion resistance tests for carbon steel and aluminum samples were carried out. The total corrosion was investigated as a loss of weight of samples. Influence of decontamination solutions on corrosion of carbon steel and aluminium was investigated by consecutive treatment of samples by the decontaminating solution for 1 hour at 25°C, and then by washing the samples with water for 15 min. The results obtained are presented in Tables XI and XII.

The surface of the steel samples was light grey, without signs of etching or pickling traces after the treatment of carbon steel samples with the modified acid decontaminating compositions No 2, 3 and 4. The corrosion deposits were removed from the surface. As the data of Table XI show, composition No 2 had the least corrosive effect, while composition 3 had the most.

Table IX. Decontaminating ability of the solutions

No.	Composition - %	Decontamination Factor (DF)	
		Carbon steel	Aluminium
3.	H ₃ PO ₄ - 10.0; HEDPA – 1.0; C ₄ H ₆ O ₆ – 3.0; SF-2U – 0.5	65.3 46.9	
2.	H ₂ SO ₄ – 9.2 H ₃ PO ₄ – 8.4 NH ₄ F – 3.0 SF-2U – 0.5	77.0 52.3 91.0	
4.	H ₂ SO ₄ – 9.2 H ₃ PO ₄ – 8.4 C ₄ H ₆ O ₆ – 3.0 SF-2U - 0.5	99.7 69.8 92.3	
10.	LABOMID-203 – 3.0	9.3 13.7 10.9	64.8 86.7 60.3
12.	HEDPA – 2.0 EDTA – 0.3 SF-2U – 2.5 C ₃ H ₈ O ₃ – 0.6	4.4 5.6 6.3	40.2 82.9 67.5
19.	Na ₃ PO ₄ – 1.0 NaOH – 1.5 (NH ₄) ₂ HPO ₄ – 0.6 SF-2U – 1.0	6.4 14.3	70.5 84.9 99.4

Comparison of the data on decontaminating ability (Table IX) and corrosion resistance (Table XI) indicates that composition No 4 has the best combination of a high decontamination ability with a small corrosion attack on carbon steel.

After the aluminium samples were treated with alkaline compositions No 10, 12 and 19, the sample surfaces were grey, dull, very etched (19); silvery, slightly dull (10) and silvery, brilliant (12). The data show that the corrosion rate for aluminium samples is in direct proportion to the alkaline content in the washing composition. So, the highest corrosion rate was noticed in the case of composition 19, containing 1.5% NaOH. The least corrosion rate was found in composition 12, which did not contain alkaline. Except of that, decontaminating ability of composition 12 was the least. At the same time composition 10 – 3% solution of standard technical washing agent “LABOMID-203” possess as a high decontaminating efficiency as well as a lower corrosion impact. Hence, this composition could be recommended for decontamination of aluminium surfaces.

Table X. Definition of detergency effect

Composition No	Sample No	Weight of clean sample (g)	Weight of dirty sample (g)	Weight of a sample after washing (g).	DE. %
19	1	55.3051	55.3250	55.3063	93.97
	1'	52.0131	52.0597	52.0144	97.21
	1''	52.0496	52.0806	52.0519	92.58
10	2	54.3787	54.4161	54.3855	81.55
	2'	50.5698	50.5903	50.5727	85.85
	2''	53.2070	53.2175	53.2078	92.38
12	3	53.2274	53.2669	53.2293	95.19
	3'	53.0618	53.0973	53.0637	94.65
	3''	51.5382	51.5564	51.5404	87.91

Table XI. Corrosion of carbon steel in decontaminating solutions

Composition No	Sample No	Sample area. (cm ²)	Weight of a sample before washing (g)	Weight of a sample after washing (g).	Corrosion rate. (g/cm ² ·h·10 ⁻⁴)
2	1	53.36	53.2160	53.1837	1.51
	2	51.97	52.4223	52.3880	1.65
	3	53.33	53.1083	53.0732	1.64
	4	51.34	51.1547	51.1254	1.43
3	5	51.76	52.0309	51.9296	4.89
	6	52.23	54.5284	54.3196	9.99
	7	54.34	45.3321	54.2642	3.12
4	8	52.69	52.6061	52.5608	2.15
	9	52.06	52.2664	52.2225	2.11

Table XII. Corrosion of aluminium in decontaminating solutions

Composition No	Sample No	Sample area. (cm ²)	Weight of a sample before washing (g)	Weight of a sample after washing (g).	Corrosion rate. (g/cm ² ·h·10 ⁻⁴)
19	1	49.19	11.3709	11.0650	15.55
	1'	49.78	11.8089	11.5860	11.20
	1''	50.32	11.9039	11.4995	20.09
10	2	49.11	11.5974	11.5903	0.36
	2'	50.22	11.9674	11.9598	0.38
	2''	49.35	11.7137	11.7071	0.33
12	3	50.14	11.9839	11.9836	0.015
	3'	51.48	12.2877	12.2868	0.04
	3''	50.57	12.0787	12.0780	0.03

2.5. Investigations of chemical stability of solid waste

2.5.1. Testing of chemical stability of foils

It is known that strippable foils based on polyvinyl alcohol are not stable in wet atmospheric conditions [3]. When these used decontaminating foils and pastes are buried in disposal sites, the atmospheric and ground waters may penetrate into them. Because of this fact the investigations on desorption of radionuclides from exhausted films were carried out.

The method of long leaching according to international and Russian standards was used [4, 5]. The method is intended for the evaluation and comparing of solid radioactive waste stability to leaching in controlled conditions. Leaching of ^{137}Cs from solid film waste in contact with distilled water during 4 months was investigated. Also, the leaching of H^+ and OH^- -ions from solid films were investigated.

The model samples for standard leaching tests were prepared using the following method. Portions of ^{137}Cs solution in HNO_3 were dropped onto Teflon trays with inner diameter 4.5 cm and dried out at room temperature. Then the contaminated trays were filled with decontaminating polymer compositions, No 1, 2, 3 and 4 (see Tables VI and VII) and were left to dry and decontaminate for 7 days. Then the dry films were extracted from trays and the linear dimensions; weights; and activity of the samples were determined.

Tubes for the leaching experiments were made from polypropylene to prevent the absorption of radionuclides on their walls. Distilled water (100 ml) was used as contact solution for leaching. The samples were placed into nylon bags and fixed in the tubes for leaching. Then the tubes were filled with distilled water. The leaching temperature was 25° and 40°C . The contact solution was replaced after 1, 2, 3, 7 days, then every week, than every month.

Concentrations of ^{137}Cs H^+ / OH^- - ions were analyzed in every solution.

The leaching rate was expressed as the normalized leach rate R_n^i for individual component i and was given by the equation:

$$R_n^i = A_i / A_0 \cdot W_0 / F \cdot t$$

where

R_n^i – is the leaching rate in grams of waste form $\cdot \text{cm}^{-2} \cdot \text{d}^{-1}$ normalized to the behavior of component i (^{137}Cs OH^- H^+);

A_i – is the amount of component i leached during the time interval;

A_0 – is the initial amount of component i in the waste form specimen;

W_0 – is the original weight of the waste-form specimen. g;

F – is the surface area of the waste-form specimen. in cm^{-2} ;

t – is the time interval of leaching. in days.

The results obtained in leaching experiments are presented in Figure 1. The results are the mean of 3 parallel experiments.

As the presented data show, the highest ^{137}Cs leaching rate was observed for film composition 1, and the lowest leaching rate for composition 2. However, on the whole the leaching rates exceeded the Russian and Belarus standards for leaching rates of ^{137}Cs from the solid waste ($^{137}\text{Cs} = 1 \times 10^{-6} \text{ g} \cdot \text{cm}^{-2} \cdot \text{d}^{-1}$).

Obviously, this could be explained by the fact that, unlike the even and equal distribution of radionuclides within a cement, bitumen or glass matrix, in the case under investigation most of the removed radionuclides were localized on the film surface and were associated with the film by the adhesion.

The investigations of polymer matrix chemical stability were carried out with the objective of evaluating and comparing different polymer decontaminating compositions. The chemical stability was monitored by measuring the contact solution pH value. The pH change caused by the leaching of H^+ and OH^- ions from the films and the pastes form parts of decontaminating compositions active agents. The pH measurements were carried out by the glass electrode in the pH-meter “pH-150”.

The data obtained are presented in Figure 2.

On the basis of the data presented in Figure 2 it is possible to determine the concentration of H^+ and OH^- - ions leached from the solid films into the contact solution and estimate the leaching rate for these ions; and therefore estimating the chemical stability of the polymer matrix. The data obtained are presented in Figure 3.

The data show that leaching rates for OH^- and H^+ - ions, ingredients of the decontaminating film matrix, are less than the leaching rates for the ^{137}Cs up to some temperature. This fact evidently shows that during the polymer film decontamination process most of the radionuclides removed from the surface under decontamination do not distribute equally in the mass of the decontaminating coating, but remains on its surface or in the upper layer of the film.

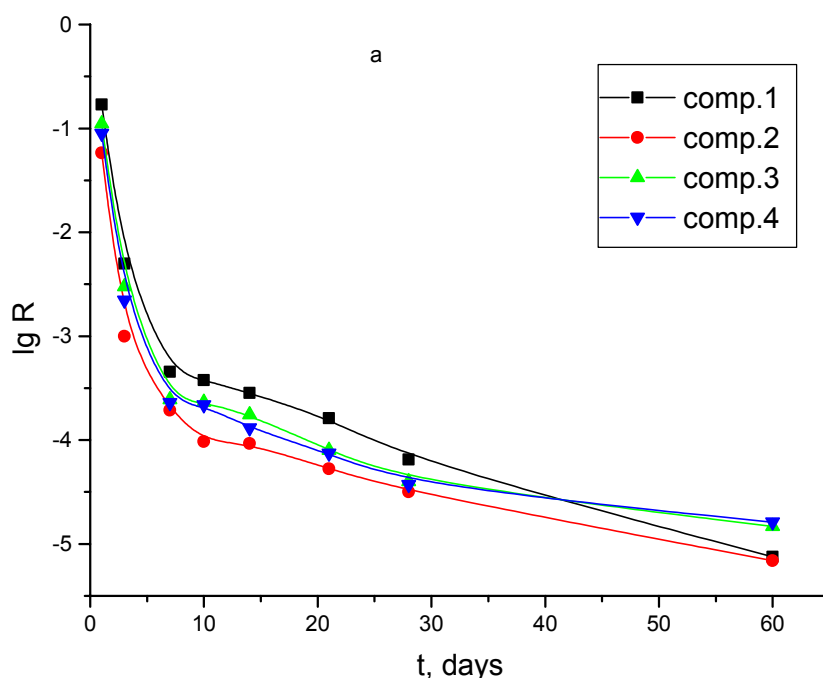


Figure 1. The leaching rate of ^{137}Cs at $25^{\circ}C$.

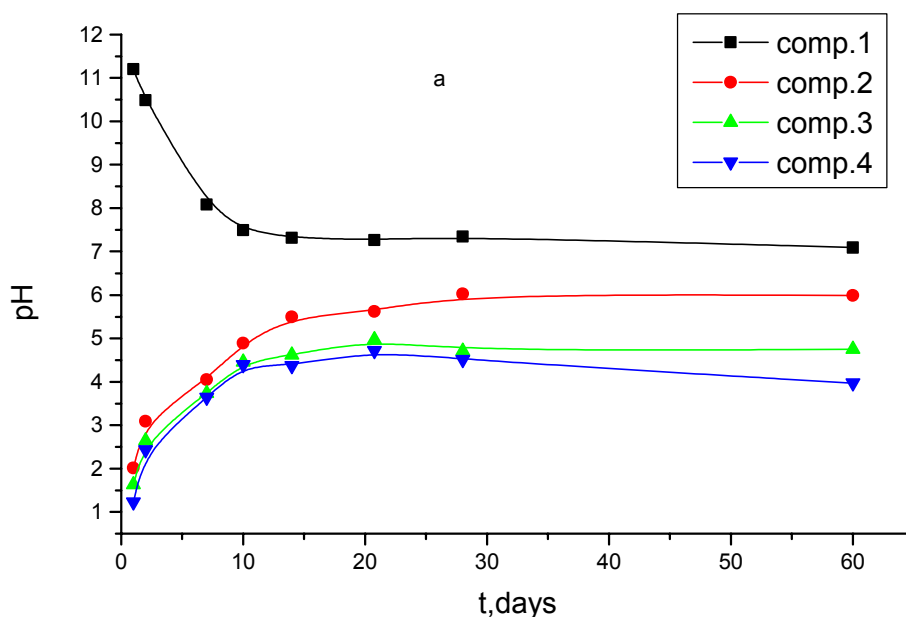


Figure 2. Changing pH of the contact solution at 25°C.

2.5.2. Testing of chemical stability of pastes

The absorption properties of strippable pastes are dependent on the chemical stability of the pastes. Leaching of ^{137}Cs from solid paste waste in contact with distilled water during two months was investigated. The leaching of H^+ and OH^- ions from solid pastes was also investigated. The results obtained in the leaching experiments at a temperature of 25°C are presented in Figures 4 and 5. The results are the mean of 3 parallel experiments.

As presented, the data show that the addition of absorbents (lignin and clinoptilolite) apparently stabilises the polymer composition. The interval of contact solution changing in the experiments on measuring of contact solution pH value was about 2 pH units in the case of pastes, and more than 4 pH units in the case of films.

At the same time, the residual paste activity after leaching is in direct proportion to the content of absorbents in the polymer composition. These data testify to the absorption properties of the pastes. The clinoptilolite is a more efficient absorbent for ^{137}Cs than lignin as seen in the data obtained and presented in Figure 6.

The solid films and coverings should be collected and treated together with another compactable radioactive waste. The quality experiments showed that the decontaminating films and coverings are combustible, do not contain corrosive active Cl^- ions and hence, could be incinerated. Combustible solid films are normally collected in transparent plastic bags. After filling, the plastic bags are removed from the bins and closed with adhesive tape.

2.6. Case study analysis

The choice of a suitable decontamination method or combination of techniques depends on a number of criteria that need special consideration. Special attention must be given to the cost-benefit analysis, which takes into account:

- (a) Availability and cost and complexity of the decontamination equipment;
- (b) The need for conditioning of the secondary waste generated;
- (c) transportation expenditures (that estimate can change very much for every concrete situation).

The integral principle of cost valuation was used for comparative analysis of decontamination costs for bath and liquid methods and for the “dry” method of decontamination via peelable films and pastes. The principle is based on the cost calculations for decontamination of 1 m² of equipment surface.

Components of the cost were depicted in physical terms: direct manpower and overhead, transportation of wastes (solid and liquid), generation of wastes (solid and liquid), liquid waste management.

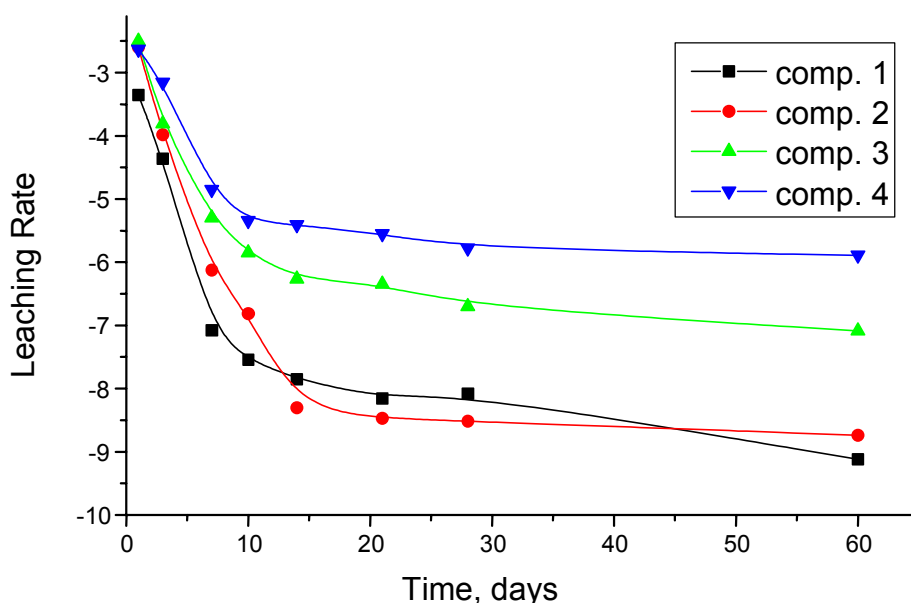


Figure 3. The leaching rates for OH ions (composition 1) and H⁺ -ions (compositions 2–4)-(log scale).

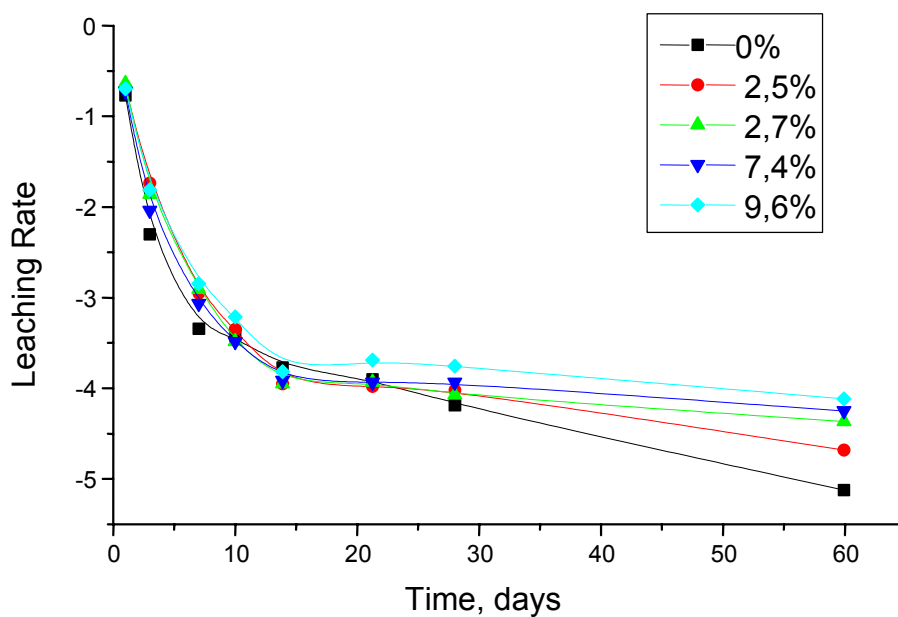


Figure 4. Dependence of ^{137}Cs leaching rate (R) on lignin content in paste (log scale).

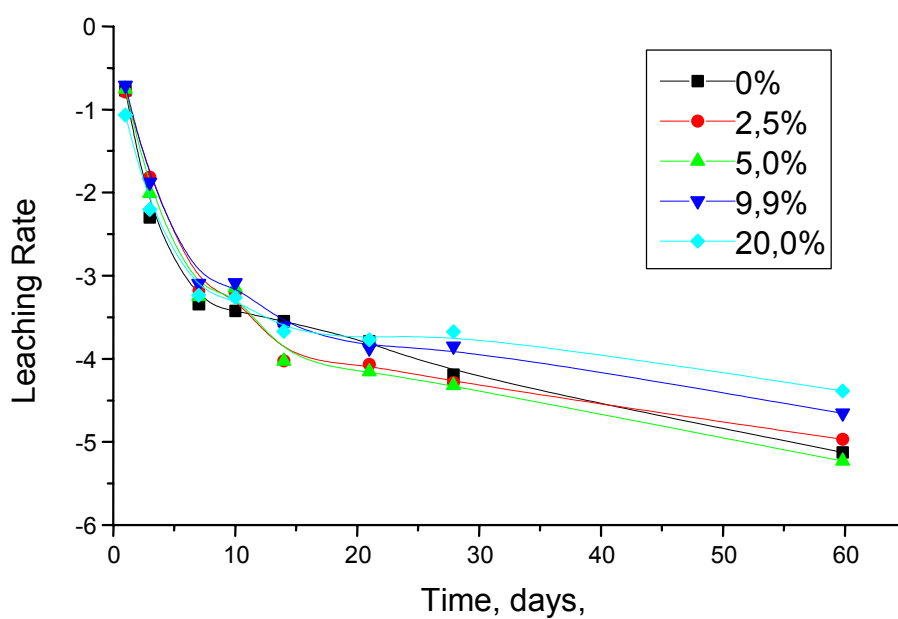


Figure 5. Dependence of ^{137}Cs leaching rate (R) on clinoptilolite content in paste (log scale).

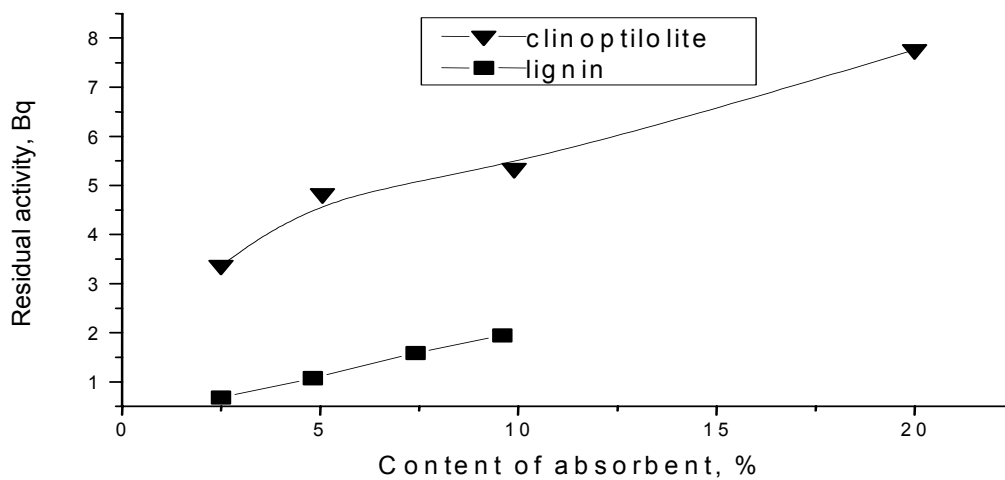


Figure 6: Dependence of residual sample activity on content of absorbents in pastes.

The cost of waste disposal was not included in the cost estimates. For methods bearing on equipment surface, the “decontamination – waste treatment” chain cost for 1 m² were composed of following different components:

- (a) manpower (cost in rubles/hours per unit of area);
- (b) consumables (cost per unit of area) — only most important consumables are considered;
- (c) overheads (% in addition to wages).
 - (i) The overheads include:
 - for fabrication of the tools.
 - the normal cost of the administration of the enterprise in charge.
 - social insurance and another Belarus overheads.

The normative financial documents of the State Specialized Enterprise for Decontamination “Polesie” were used as initial data for comparative analysis. The calculations were done on the basis of Belarus regulations, economic infrastructure, and costs. The data obtained are presented in Table XIII.

As the data presented show, the cost of liquid waste cementation composes about 66% of the decontamination cost for the liquid bath method. At the same time the cost of the technological chain “decontamination-waste treatment” for the dry decontamination process is 2.5 times lower then the cost of liquid method of decontamination.

Table XIII. The cost* of decontaminating 1 m² surface, including waste management

Cost Groups	Bath method	Peelable coatings		
		Paste 1 with lignin	Paste 2 with clinoptylolit	Film
Consumables	17.04	20.98	16.12	10.19
Decontamination	180.25	135.19	128.72	120.83
Waste. kg/m ²	3.0 (liquid)	0.3	0.4	0.05
Cementation of liquid waste	118.98	-	-	-
Transportation. loading/unloading	17.35	-	-	-
Complete cost of the cycle “decontamination – waste management”	316.58	156.20	144.88	131.07

* costs in thousands of Belarus rubles.

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Decommissioning of a small reactor (BR3 reactor, Belgium)

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Abstract. Since 1989, SCK•CEN has been dismantling its PWR reactor BR3 (Belgian Reactor N°3). After gaining a great deal of experience in remote dismantling of highly radioactive components during the actual dismantling of the two sets of internals, the BR3 team completed the cutting of its reactor pressure vessel (RPV). During the feasibility phase of the RPV dismantling, a decision was made to cut it under water in the refuelling pool of the plant, after having removed it from its cavity. The RPV was cut into segments using a milling cutter and a bandsaw machine. These mechanical techniques have shown their ability for this kind of operations. Prior to the segmentation, the thermal insulation situated around the RPV was remotely removed and disposed of. The paper will describe all these operations. The BR3 decommissioning activities also include the dismantling of contaminated loops and equipments. After a careful sorting of the pieces, optimized management routes are selected in order to minimize the final amount of radioactive waste to be disposed of. Some development of different methods of decontamination were carried out: abrasive blasting (or sand blasting), chemical decontamination (Oxidizing-Reducing process using Cerium). The main goal of the decontamination program is to recycle most of the metallic materials either in the nuclear world or in the industrial world by reaching the respective recycling or clearance level. Overall the decommissioning of the BR3 reactor has shown the feasibility of performing such a project in a safe and economical way. Moreover, BR3 has developed methodologies and decontamination processes to economically reduce the amount of radwaste produced.

1. Introduction: The BR3 decommissioning summary

The BR3 reactor was the first pressurized water reactor (PWR) installed and operated in Europe. While its rated power level is low (40 MW(th), 10.5 MW(e) net), it contains all the features of commercial PWR power plants. The reactor was used at the beginning of its lifetime as a training facility for future NPP operators. Later on it was also used as a test bench, in full PWR conditions, for new types of nuclear fuel (e.g. MOX, consumable poison, high burnup,).

The reactor was shut down in 1987 after 25 years of operation.

In 1989, BR3 was selected by the European Union as one of four pilot dismantling projects, included in the third EU five-year research programme on decommissioning of nuclear installations. The project started in 1989 and is ongoing. The first part of the pilot project (1989–1994) involved the decontamination of the primary loop and the dismantling of all the highly radioactive reactor internals.

In 1994, an extension of the contract was signed with the European Union, covering the dismantling of the first set of reactor internal components, which were removed from the reactor 30 years ago. The main goal of this contract was to allow the comparison of an immediate dismantling operation with a deferred operation after a 30-year cooling period.

In 1996, it was decided to carry on with the dismantling of the BR3 reactor pressure vessel (RPV). The first technical acts of this important project were executed at the end of

1997. The dismantling of the RPV was also part of a European contract. In Summer 2000, the last cut on the RPV was carried out and the cut pieces are now in the process of being transferred to the storage facility.

In 1999, the MEDOC decontamination workshop went into industrial service with material clearance as the main objective.

Work is now concentrated on the dismantling of the primary circuit and its large components which will be cut using a promising and quite new cutting technique: High pressure water jet cutting.

Carrying out an important part of this project by itself, the SCK•CEN has gained important experience which allows it to be a specialist in cost evaluation, strategy, study, remote cutting techniques, decontamination techniques, waste management and ALARA evaluations.

2. The dismantling of the reactor pressure vessel

A detailed study for the complete dismantling of the RPV, either in air or under water was carried out. Based on the results of the preceding projects, the mechanical cutting processes were first promoted and analysed.

The studies assessed the overall manpower requirements, scheduling and costs of both operations. For the dismantling of the RPV, the underwater method was finally selected. The RPV was surrounded by an annular Neutron Shield Tank (NST), allowing the vessel to be submerged with only the three penetrations for the primary loop piping needing to be sealed to assure the leak tightness of the pool during the operation.

Further study of the RPV dismantling problems led to the analysis of two different approaches: the in situ dismantling, where the RPV remains in place (under the bottom of the refuelling pool) while it is cut into rings, and the "one-piece removal", where the vessel is removed in one piece into the refuelling pool, and then segmented into pieces ready for packaging.

The advantage of the latter is the accessibility of the RPV and its insulation shroud from the outside, providing the possibility for reuse of the dismantling tools and equipments designed for the internals dismantling. Moreover, this approach greatly simplifies the dismantling of the RPV insulation shroud situated at about 100 mm outside the vessel wall.

2.1. Preliminary operations

These operations (see Figures 1 & 2) were executed with a dry refuelling pool, the RPV still located in its cavity under the bottom of the refuelling pool. Access to the pool floor was possible but had to be reduced as much as possible for radiation protection reasons.

2.1.1. Separation of the RPV from the bottom of the reactor pool

The selected process for cutting at the bottom of the reactor pool was the plasma arc torch handled by an operator. The cutting has to be done quickly in order to limit the dose to the operators. In addition to this operation, different cuts at the bottom of the reactor pool were also needed to give access to the fastening bolts of the RPV support flange, to give

access to the hot and cold leg thermal insulation and to allow the installation of the sealing equipment necessary for the future watertightness of the pool.

2.1.2. Removal of the asbestos situated around the primary pipes near the RPV

This operation was carried out by SCK•CEN personnel, as the nuclear hazard was estimated to be far above the asbestos hazard. Nevertheless, to avoid the spread of asbestos fibers, a double confinement was installed in the RPV pool.

2.1.3. Separation of the RPV from the hot and the cold legs

Cutting of the primary pipes at the outside of the bioshield

The main concern for this operation was the cutting of the pipes at the RPV flange level. Because of the very tight space available to perform this operation, access was needed through the primary pipes on the bioshield side. This operation was carried out with a common type of automatic pipe cutter, using two lathe tools diametrically opposed.

Cutting the primary pipes near the RPV

This operation was delicate due to the fact that access was only available from the inside of the piping. With the help of an industrial partner, an automatic milling cutter able to cut the necessary thickness was developed. The challenge was to have a machine fitting into a diameter of 254 mm, able to cut up to 110 mm wall thickness. Finally, it was decided to make a second cut of the primary pipe connections just above the support flange of the RPV in order to gain access to all the RPV fastening bolts. The cutting tool is an automatic milling cutter with a diameter of 30 mm for the first part of the cut, 25 mm for the second, deepest, part.

2.1.4. Separation of the RPV from the NST

A pneumatic wrench was selected for the removal of the 24 bolts fastening the RPV to the Neutron Shield Tank. This operation took about three times longer than foreseen due to the high level of corrosion.

2.1.5. Reinstallation of the water tightness of the NST and the reactor pool

As the RPV and its primary pipes were part of the pool leak tightness system, the openings left after cutting the primary piping, located in a very tight space had to be sealed. The operation was carried out with an industrial partner, who developed a system based on an epoxy-based polymer and a form-shaped sealing system. Cold testing was carried out on a full scale mock up and everything was ready for the installation. During the installation, a major positioning problem arose which will be discussed further on.

Finally the RPV was ready to be lifted. A guiding system had been also installed as the mechanical clearance between the RPV and the sealing devices was less than 10 mm. On August 24, 1999 the pressure vessel (28 ton) was lifted up in one day, using a new gantry crane installed above the RPV pool. The water level in the pool was raised at the same pace as the RPV lifting.

2.1.6. Removal of the insulation shell

The insulation shell is bolted to the RPV through a series of bolts on the four quadrants of the RPV and on the upper side it is bolted to the RPV supporting skirt. It was necessary to remove 60 bolts to free the insulation shell from the RPV. Because of the horizontal position of these bolts they had to be drilled by a remote hydraulic hole cutter. In order to easily reach the different levels at which the bolts were placed, the remote hydraulic hole cutter could move up and down along a beam. Here again, mock-up tests were used.

During the execution of this dismantling task, two problems were encountered. First of all, there was a positioning problem for the cutting tool and second, there was a visibility problem with the pool water. These will be discussed further on in this paper.

2.1.7. Removal of the insulation and the fastening profiles of the insulation shell

The insulation shell was bolted on the RPV by T-shaped fasteners and connection pieces on two levels. Between and on top of these fasteners there is fibreglass thermal insulation, fastened with a metal mesh. The insulation was also held together with metal straps. On the bottom side of the RPV the insulation is held against the RPV with eight straps. These straps are attached on the RPV by bolts through the insulation material.

As the mesh was totally rusted, the removal of the insulation was done using a long handling tool. The liberated insulation fell into a fishing net previously installed on the floor of pool. By remotely closing the fishing net, the insulation was taken out of the water and removed as standard low level waste.

The T-shaped fasteners on the insulation shell were the last items to be removed before the actual cutting up of the vessel could begin. The first approach was to unscrew the bolts of these fasteners. However, due to heavy corrosion of the bolts, these fasteners could easily be ripped off the vessel with a hook. As the fasteners were not very active, their further dismantling was done by hand.

2.2. Dismantling operations of the RPV

The chosen dismantling strategy reuses the existing circular saw (and the existing band saw). New tests were necessary to define new machine cutting parameters because compared to the previous phase of the project (i.e. dismantling of the reactor internals), a different sort of base material (carbon steel instead of stainless steel) and different thickness (112 mm instead of 25 mm) had to be cut.

As the cutting equipment was contaminated during previous phases of the project, the mock-up tests to prepare for the operation had to be carried out in the controlled area.

2.2.1. The horizontal cutting of the RPV using the circular saw

A new clamping device was designed, primarily for use during the first cut with the circular saw. The design of the clamp allowed the upper part of the RPV to be held during and after the cut. The lower part of the RPV was difficult to clamp due to its spherical shape (see Figure 3).

A mock-up of the reactor pressure vessel was made to carry out tests of the cutting technique with the circular saw.

The first series of mock-up tests brought some problems to light:

- the mock-up vibrated a lot;
- the selected type of sawblade seemed not ideal for the purpose.

The results of this first series of tests led to:

- the construction of a stiffer clamping system;
- the detailed analysis of the requirements of the cut, concerning the saw blade design;
- the organization of an extended second series of tests.

The second series of tests had as objectives:

- to validate the new clamping system device (see Figure 4);
- to try and optimize the cutting sequences and their associated parameters.

Both objectives were finally reached and the actual work could start. The estimated duration of the operation was 107 shifts on the basis of a mean feed speed of 15 mm/min. In fact the actual duration of the operation was 65 shifts, a reduction of 42 shifts, mainly because of much better performance of the sawblades (feed speed up to 80 mm/min and a longer life span). Similarly, the actual integrated dose received was only 3.652 man-mSv in place of the 9.784 man-mSv estimated. Figure 5 shows the refuelling pool during horizontal cutting.

2.2.2. The vertical cutting of the RPV using the band saw machine

Vertical cuts through the nominal thickness of the RPV (112 mm)

During mock-up tests, a few cuts were carried out in the SS clad carbon-steel wall of the RPV mock-up without any problem, using a feed speed of about 20 mm/min. This immediate success was thanks to a similar job that the BR3 team had already carried out during the previous phase of the project (i.e. the dismantling of the instrumentation collar). There were no major problems with the cutting operation. However, the cutting speed used was much lower than during the mock-up tests: 9,5 mm/min. During the segmentation of the two first rings, many sawblades were used. To avoid this high consumption of sawblades for the other rings (included the additional time required for the blade exchange), the cutting speed was reduced.

Vertical cuts directly through the RPV flange followed by a horizontal cut through vessel-insulation-shroud (all 3 in the same time)

The cuts through the flange (thickness 355 mm) were successful at 7 mm/min. Special attention had to be paid when cutting near the weld situated below the flange where there is a risk of jamming the blade. The major difficulty when cutting the RPV flange was due to a 2 cm thick supporting shroud under the flange. This meant that when cutting the upper part of the RPV, the band saw had to simultaneously cut the flange followed by the vessel + insulation + shroud. The influence of the insulation on the cutting performances was unknown and had to be tested. Finally, the tests and the actual cuts were carried out without any problem. Figure 6 shows one piece of the reactor pressure vessel. Since this piece had only a low level of activity, it was put in the drum "hands-on". The pieces closer to the core, much more activated, were manipulated remotely under water.

3. Encountered problems and solutions

During the dismantling phase of the RPV, the team encountered two major problems: some "non conformities" or discrepancies with the "as built" drawings and severe turbidity problems with the cutting pool.

3.1. Problems with the "non conformities" of as built drawings

As already explained earlier in the text, one had to retain the watertightness of the reactor pool. This would be done with three special designed sealing devices. Early on in the dismantling process one had to stop the operation because it was impossible to properly position the sealing devices due to discrepancies between the "as built" drawings and the actual equipment. Figure 7 illustrates the problem: a metallic lath (which was thicker than indicated) made it impossible to push the sealing device correctly against the NST inside wall. Therefore, the design of the sealing devices had to be revised, and the sealing devices themselves had to be adapted. The positioning of these devices was finally carried out in June 1999 instead of March 1999.

Discrepancies are a common problem in the dismantling of old nuclear facilities. Another problem in the same category is the one encountered with the insulation shell removal. At the outset, it became almost impossible to locate the screw heads due to a high level of corrosion on the shroud surface (see Figure 8). Therefore it was impossible to locate these bolts to cut them with a hole cutter machine. It was then decided to cut the entire circumference of the core shroud at the corresponding level of the bolts. This method required 10 times more holes to be cut than planned.

3.2. The turbidity problem of the pool

When the insulation shell (a protective metal sheet for the thermal insulation situated around the RPV – see Figure 8) was removed a major problem occurred: significant water turbidity appeared. This was due to the thermal insulation which became breakable into something looking like dust but also to rust. Sometimes, the visibility was so bad that the operation had to be stopped. Additional filtration and purification facilities were installed to solve this problem. The same problem appeared again when the cut with the bandsaw for the removal of the vessel flange was carried out. Remaining insulation, situated under the vessel flange again entered the pool water and caused a new turbidity problem. Figure 9 shows the turntable coming out of the pool water.

To clean and purify water, it is necessary to first remove the particles in suspension and then the dissolved ions in order to lower the conductivity. Filtration tests were performed at the pilot scale, which showed that it was necessary to use at least a 1.2 μm filter.

The existing refuelling pool water purification system comprises a filter unit with a capacity of about 20 m^3/h and a 210 l ion-exchange column with a capacity of 2.5 m^3/h . The filters used are 10" wound 1 μm polypropylene filter cartridges. The ion exchanger is a homogenous mixture of strongly acidic cation resin and a strongly basic anion resin.

This system was insufficient to deal with the heavy pollution observed; moreover it appeared that the ion exchange column was saturated. Therefore, several actions were undertaken:

- installation of an additional mobile filtration unit with a capacity of 20 m³/h pumping directly in the pool
- replacement of the saturated resins
- installation of two mobile ion exchange columns at the outlet of the mobile filtration unit with a capacity of 2 to 3 m³/h.

After these items were completed, the visibility could be kept under control except during short periods of peak pollution corresponding to release or resuspension of rust or fibres. To maintain a high water quality, it was necessary to regularly replace the saturated filters or resins. During the reactor pressure vessel cutting process, the production of secondary waste amounted to about 2.9 m³ of burnable cartridges and 0.7 m³ of burnable ion exchange resins.

4. RPV waste

The Belgian National RadWaste Authority (ONDRAF/NIRAS) has the responsibility of establishing the different acceptance criteria for waste types and waste packages. For the solid waste (big pieces), there are three major groups of waste, distinguished by the contact dose rate. These are Low-Level solid waste (LLW) with a contact dose rate < 2 mSv/h, Medium-Level solid waste (MLW) with a contact dose rate between 2 mSv/h and 0.2 Sv/h and High-Level solid waste (HLW) with a contact dose rate > 0.2 Sv/h.

There are only two different types of waste packages, namely the standard 400 l drum and the standard 200 l drum. The 400 l drum is used for large pieces. At the waste facility these drums will be filled with concrete. The 200 l drum is used for small pieces and is intended for supercompaction.

The reactor pressure vessel itself led to the production of a high volume of waste, more particularly high- and medium-level waste. For radiation protection reasons, this waste had to be manipulated under water. In total, nine shipments were made to the Belgian waste conditioner and intermediate storage facility representing a volume of 3.6 m³ of high-level waste (including highly activated swarfs). The medium-level waste was manipulated with the same rack system and represented a volume of 4.8 m³. Low-level waste, primarily the vessel flange and the bottom ring led to a volume of 6.8 m³.

5. Management routes for contaminated metals

5.1. Disposal routes

Dismantling of a nuclear reactor produces large quantities of materials and associated gaseous, liquid and solid effluents. Not only primary materials are produced i.e. the items dismantled but also secondary materials e.g. tools, equipments, new hardware for dismantling and decontamination and secondary effluents from the dismantling operations.

The major solid materials coming from the dismantling operations are:

- Burnable wastes
- Low to High level massive metallic wastes
- Low to High level super-compressible metallic wastes
- Massive concrete wastes
- Concrete and bricks of super-compressible rubble

- Sludge
- Various light non metallic super-compressible materials
- Special waste

Three main material categories can be distinguished:

- (a) Material which can be considered as conventional and treated as such, e.g. disposed of as industrial waste or recycled in the industry: emergency power supply, tertiary loop, components outside the controlled area.
- (b) Material which has to be disposed of as radioactive waste, e.g. activated materials or heavily contaminated material which cannot be technically or economically decontaminated or cannot be recycled or re-used: reactor pressure vessel and its internals, highly activated concrete, contaminated materials, etc.
- (c) Material which has to be considered as radioactive material, but as an alternative to its disposal as radioactive waste can be cleared unconditionally after decontamination, cleared after melting or recycled in the nuclear industry: contaminated piping, reservoirs, pumps, structural equipments, contaminated concrete, etc.

5.2. Work organization

The dismantling of a nuclear facility is a complex task. Therefore the dismantling operations are divided in hundreds of different tasks or work packages. For each task, a working procedure is established. This procedure gives the details of the work to be done and makes an analysis of the safety aspects (conventional and radiological). The work is only started after approval from the Health Physics group attached to the facility.

The main steps followed for a typical dismantling work package, such as the cutting of a contaminated loop, are:

- On site **dismantling** in large pieces
- **Cutting** in small pieces in a ventilated workshop
- **Sorting**
- **Identification**
- **Temporary storage**
- **Treatment** (washing, chemical decontamination)
- **Characterization**
- **Disposal.**

In this process, the crucial point is the **sorting**. It has to be carried out as soon as possible after dismantling (cutting) in order to guarantee the traceability i.e. where does it come from, what is its history? The sorting of the material must be well prepared in advance to accelerate the operation. The operator must know the destination of the material. The decision depends on the contamination level, the geometry of the pieces, the materials composition, the nature of the contamination, etc.

The sorting of the dismantled material leads to the creation of "batches", groups of materials that will follow the same disposal routes. Every batch carries a unique identification label. The content of a batch, its status and its location must be known at each moment.

All relevant information is collected:

- A unique identification number is written on a label fixed on the batch; this label gives the content of the batch, its weight and the disposal route selected.
- The actual status is reported in the database
 - In buffer storage before treatment
 - In the characterization process
 - Disposal route selected
 - Cleared, disposed of as radwaste, or in storage
- Finally, a document is created with all the necessary approvals. It functions in the selected disposal route as a clearance document, a request for treatment as radioactive waste or an authorization to send to a melting facility.

5.3. Treatment of radioactive metals by melting

Nowadays, "nuclear" melting facilities are in operation in several countries for the treatment of low-level metallic wastes. To be cost effective, these installations must have a sufficient throughput. At the moment, Belgium does not have an available facility so contracts were negotiated with facilities abroad.

5.3.1. Melting for recycling in the nuclear world

Low level radioactive materials may be recycled in the nuclear world. The melted materials are used for the fabrication of shield blocks or for the fabrication of radioactive waste containers. SCK•CEN has an agreement with GTS-Duratek in the USA; the recycled materials are used as shielding for the DOE facilities. The materials meet stringent composition and radiochemical criteria. The secondary wastes are conditioned and disposed of by Duratek.

Future shipments of materials in this category are being considered such as:

- Materials slightly activated: metal shielding, pool liners, fuel storage racks.
- Materials of complex geometry not possible to decontaminate economically: heat exchangers, pumps, complex structural materials, small pipes.

5.3.2. Melting for clearance

Some dismantled materials are either very low contaminated, very difficult to measure or not homogeneously contaminated. For these materials, it can be advantageous to send them to a nuclear foundry. Melting offers several advantages:

- It decontaminates the metals by volatilization of some nuclides (e.g. ^{137}Cs) or by transfer to the slag (e.g. heavy nuclides such as alpha emitters).
- It allows an accurate determination of the radionuclides content thanks to the homogeneity of the metal melt.
- The amount of secondary waste (dust, slag) is rather low.

Future shipments in this category are being considered, including:

- Materials not able to meet the criteria for direct clearance after decontamination.
- Heterogeneous materials containing hot spots and/or activity difficult to measure.

The materials will be separated by type (carbon steel, stainless steel, copper, aluminum); lead and galvanized steel are not accepted in this foundry. The paint must be removed from the pieces either by sand blasting in our facility or by sand blasting in the Studsvik facility. The presence of organic matter and encapsulated water must also be avoided.

5.4. Clearance of metallic materials

The steady increase of conditioning and disposal costs as well as environmental concerns and public perception are pushing the nuclear sector to decrease the amount of radioactive waste generated and hence produces a strong incentive for the development of thorough decontamination processes and procedures for the clearance of obsolete radioactive materials and their reuse in the industrial sector or their disposal as industrial waste.

The clearance of radioactive materials requires a combination of factors to be successful:

- Procedures and well-defined clearance criteria: a consensus is not yet achieved on international level and generally a case by case management is still applied. IAEA, EU, OECD are progressively converging towards some harmonization. The council Directive 96/29 Euratom, that had to be implemented in national legislation by May 2000, does not prescribe the application of clearance levels by competent authorities. It is up to the Competent Authorities to establish clearance levels below which the disposal, recycling or reuse of materials is released from the requirements of the Directive. In our case, the Health Physics department under supervision of the Competent Authority establishes procedures. This procedure is still a "case by case" practice and is applied currently for the clearance of materials from the BR3 dismantling.
- A strict accounting of the dismantled materials comprising origin of the materials, treatment performed and characterization results.
- The traceability of the materials must be guaranteed at each step: this can only be achieved with a strong Quality Assurance program, presently being implemented.

The characterization of materials to be cleared is still a difficult topic. Materials, which are candidate for clearance without melting, can be subdivided into 3 categories:

- Materials of simple geometry for which a 100% surface measurement is possible using hand held β monitors. For these materials, surface specific clearance values are established and the procedures are well known. The values used are 0.4 Bq/cm^2 for $\beta\gamma$ emitters and 0.04 Bq/cm^2 for α emitters.
- Homogeneous materials such as concrete rubble for which only volume or mass measurement is possible. For these materials, international mass specific guidelines are generally followed and measurement procedures are available (e.g. γ spectrometry of the whole amount in a 200 l drum or statistical sampling after homogenization). There are for the moment no fixed legal values for the clearance of such bulk materials; the health physics consider this still on a case by case basis. Their decision depends not only on the measured level but also on the origin of the material, its history and its final destination (e.g. recycling as scrap materials or disposal as industrial waste).
- Materials of complex geometry and/or heterogeneous (pipes internally contaminated, pumps, valves.): the question is how to prove that the activity level is lower than the current clearance guidelines? A procedure, based on a double measurement method has been worked out.

We use:

- Hand held β monitors for direct surface measurements
- For volumetric measurements
 - Spectroscopy HPGe detectors: Q2-220 l waste barrels.
 - Versatile spectrometry with HPGe detectors: Isocs system.
 - Gross gamma counting with scintillation detectors: the ESM CCM monitor.

The procedures followed are:

- Hand held monitors for easy to measure materials; 100% of the surface measured twice at a max 3 months interval for materials submitted to a decontamination treatment (sweeping effect).
- For homogeneous materials, we actually use the Q2 spectrometer for measurements of 200 l drums.
- For heterogeneous materials, we have two possibilities:
 - The materials are sent to a nuclear foundry, which allows a further decontamination and a reliable measurement thanks to the homogenization.
 - We combine two measurements techniques:
 - (i) A gross gamma counting with scintillation detectors for measurements of individual pieces or of small batches (1/10 of a 200-l drum).
 - (ii) A Q2 spectrometer for the determination of the specific activity per individual gamma nuclide.

The Q2 spectrometer is well known whereas the ESM CCM gross gamma counter for the small batches as well as the Isocs system were recently developed and the results obtained were compared.

6. Decontamination techniques

For metals, we use mainly:

- Manual washing or cleaning in an ultrasonic rinsing bath: mainly for pieces only slightly contaminated by deposition of contamination on external surfaces (demineralized water piping, structural pieces, instrumentation boxes...).
- Wet abrasive decontamination: mainly used for rusted or painted pieces of simple geometry in which the contamination is fixed in the oxide layer or in the paint (structural equipment, beams...). An installation called ZOE is used for the treatment of pieces up to 3 t and 3 m long maximum.
- Hard chemical decontamination with the MEDOC Cerium process: mainly used for stainless steel pieces heavily contaminated up to 20,000 Bq/cm² ⁶⁰Co (primary loop, tanks,...). The Medoc installation has a capacity of about 0.5 to 1 t of metals per batch which can be treated in one day.

Up to now, about 50 tons of metals have been treated in these different decontamination workshops. About 10 to 20% were not directly cleared; they are then sent to a nuclear melting facility for further decontamination and clearance or for recycling in the nuclear industry; the choice between the melting facilities is a function of the residual contamination present.

7. Conclusions

The BR3 Pilot Dismantling Project has allowed various dismantling techniques to be tested under fully representative conditions. The dismantling of the highly radioactive internals allowed the comparison of different cutting techniques and demonstrated the feasibility of such operations. Likewise, a reactor pressure vessel was totally removed from the plant containment after safe segmentation.

The comparison of the techniques led to a preference for mechanical segmentation techniques, which are well known in the industry and require only adaptation for working under water and in a nuclear environment.

The experience gained with a pilot reactor like BR3 provides SCK CEN with the knowledge of detailed costs, doses, waste and risks for the dismantling of a nuclear reactor. The SCK•CEN is now able to evaluate the cost and duration of such an operation, to give advice and to support dismantling operations, and to advise on design and operational guidance in order to facilitate future decommissioning of nuclear installations.

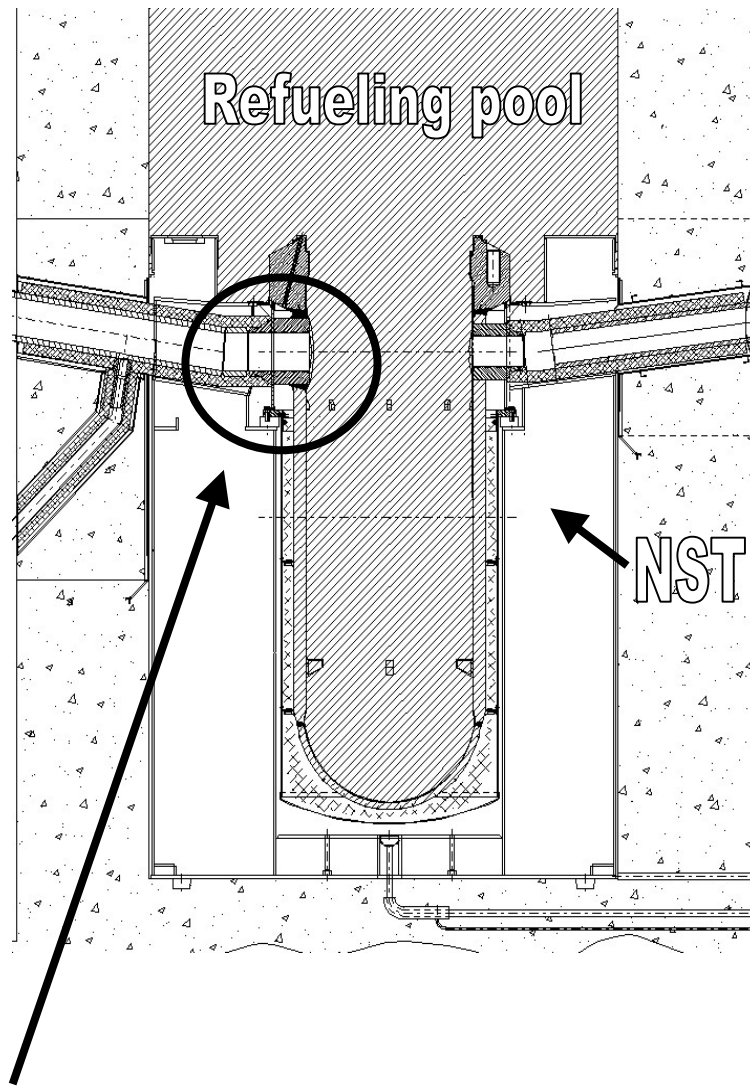
The management of dismantling materials, with the objective of minimization of the amount of radioactive waste by applying decontamination and clearance or recycling, is a complex task due to the high variety of materials, the high variety of contamination levels and the low level measurement issues.

We have demonstrated that this is technically feasible and that it is cost effective since the overall cost of the decontamination-recycling-reuse route is still lower than the disposal and replacement route. Moreover, it saves natural resources and decreases the radioactive waste volumes.

This choice implies the setup of a strong Quality Assurance program to guarantee the traceability and to push the industry to develop cost-effective decontamination and measurements techniques.

Harmonization of the different regulations and the adoption of "reasonable" clearance levels are major efforts that must still be carried out.

Even though BR3 was a small power plant, the results and lessons learned can be used to derive data for other nuclear installation dismantling: the radiological, waste and technical problems are similar.



Look to figure 2 for close-up

Figure 1. Due to the presence of the NST, there is no "easy" access to the thermal insulation of the RPV.

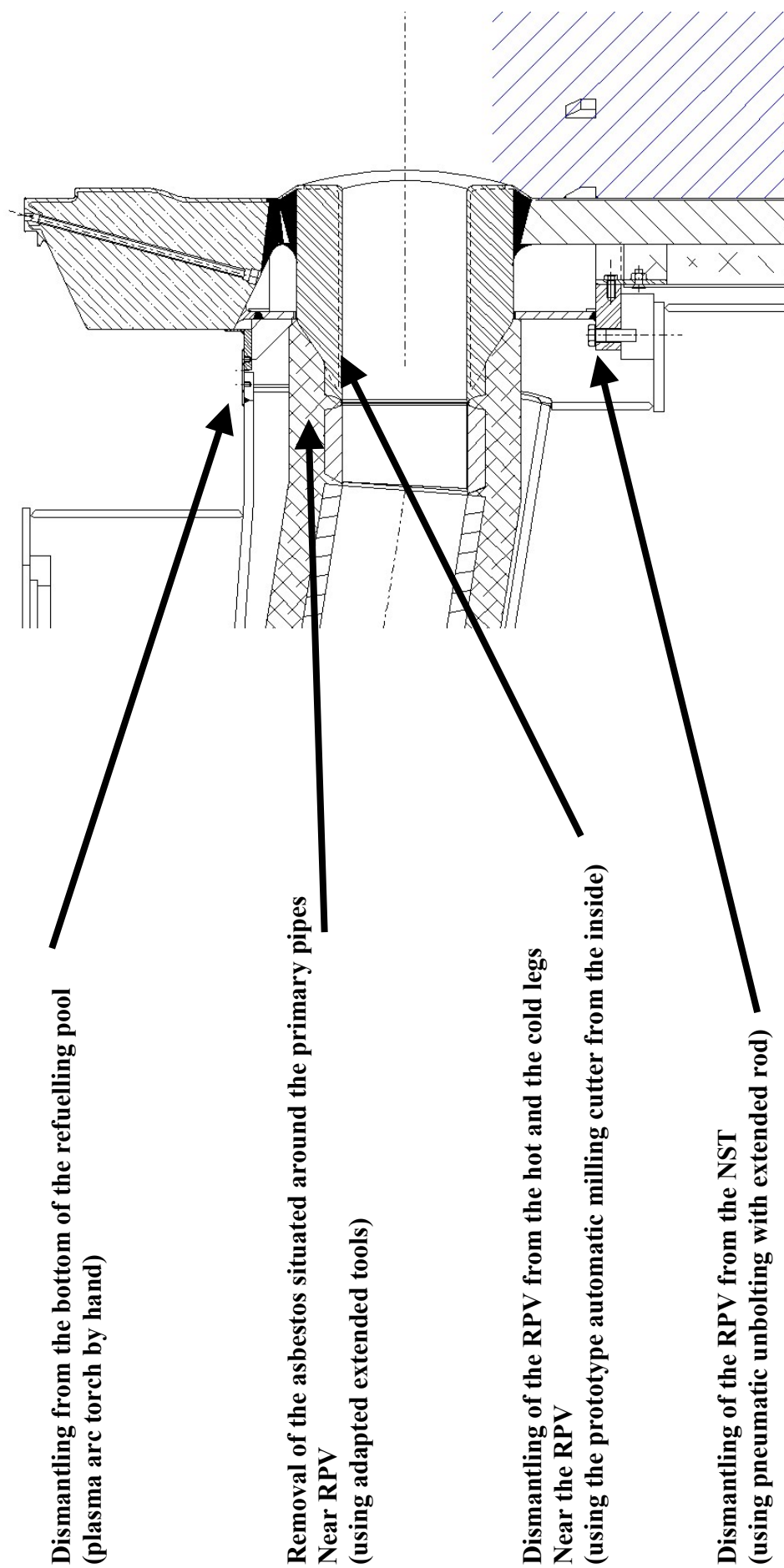


Figure 2. The dismantling steps for the reactor pressure vessel.

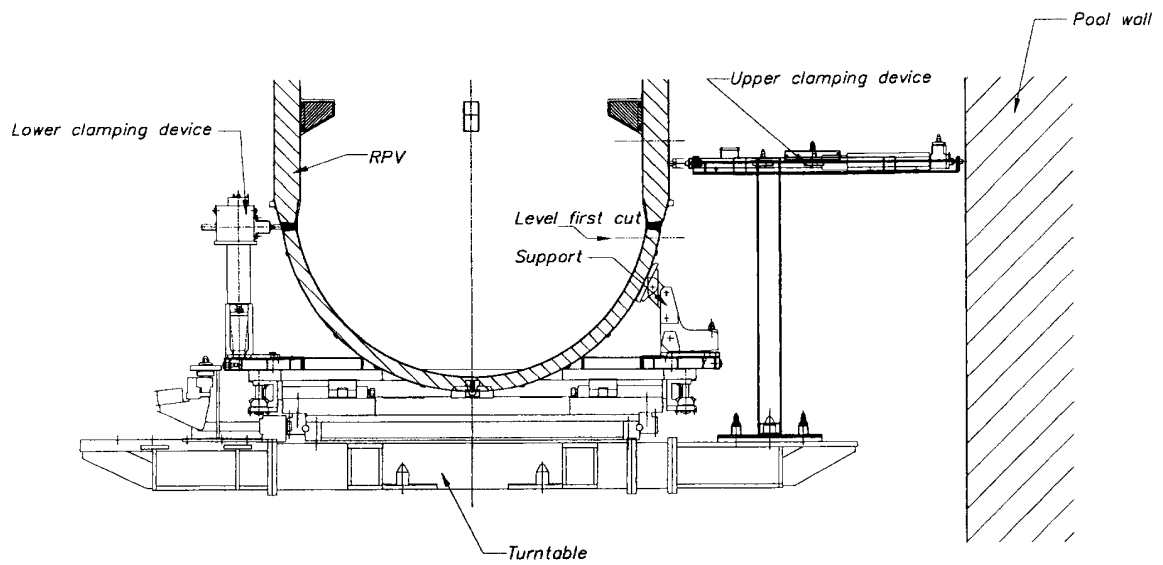


Figure 3. The spherical shape of the RPV bottom required additional clamping devices.

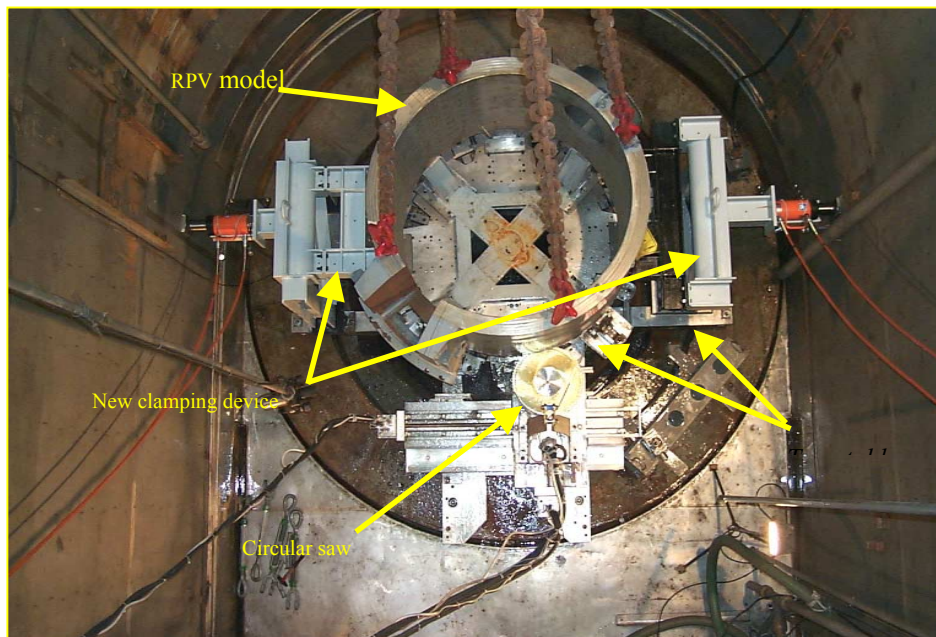


Figure 4. New clamping devices were built.

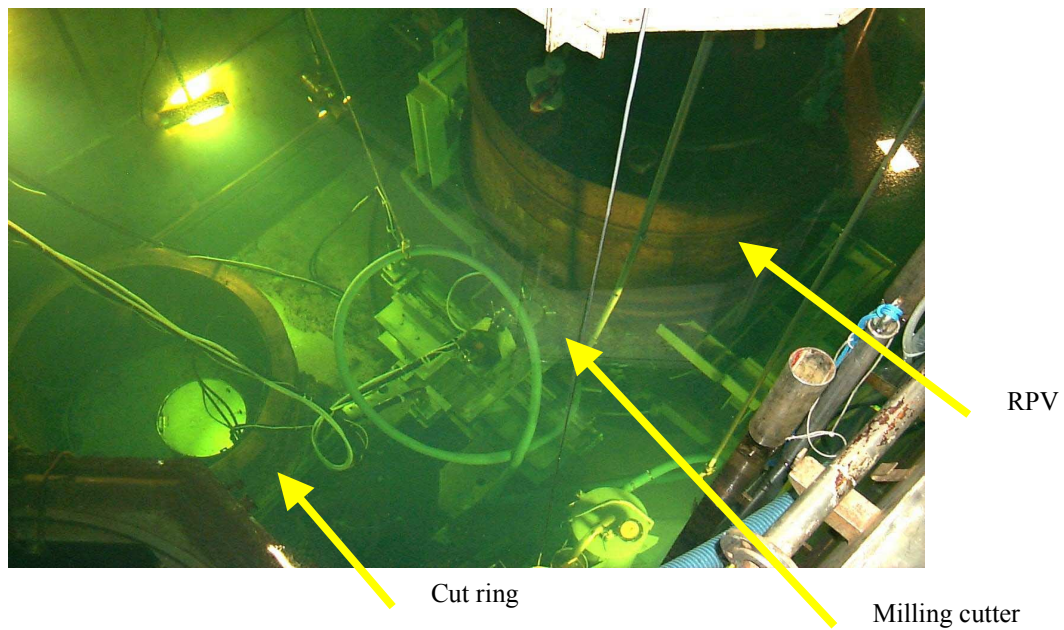


Figure 5. Dismantling activities during the horizontal cutting.

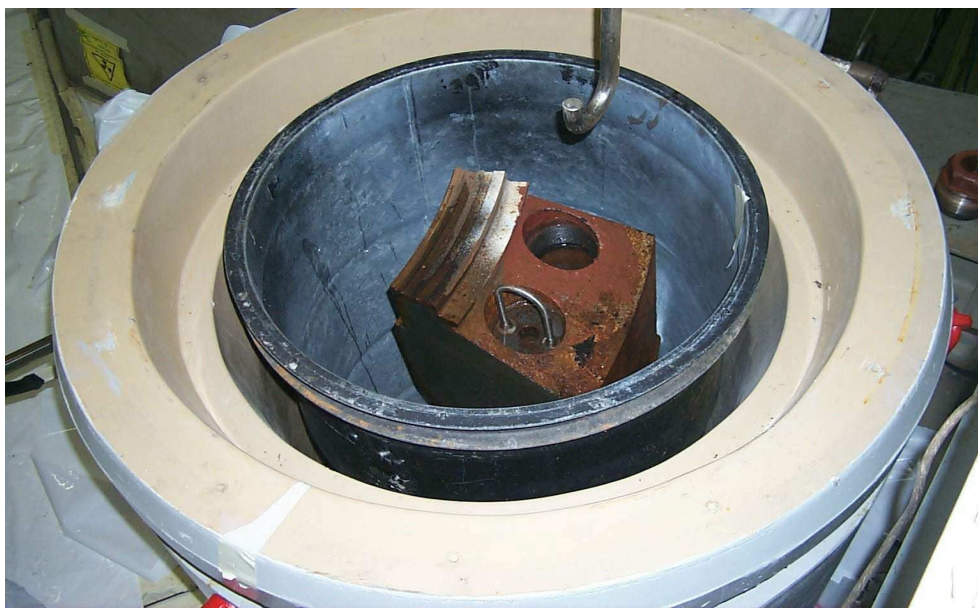


Figure 6. One piece of the RPV flange put into a 400 l drum for disposal as low level waste.

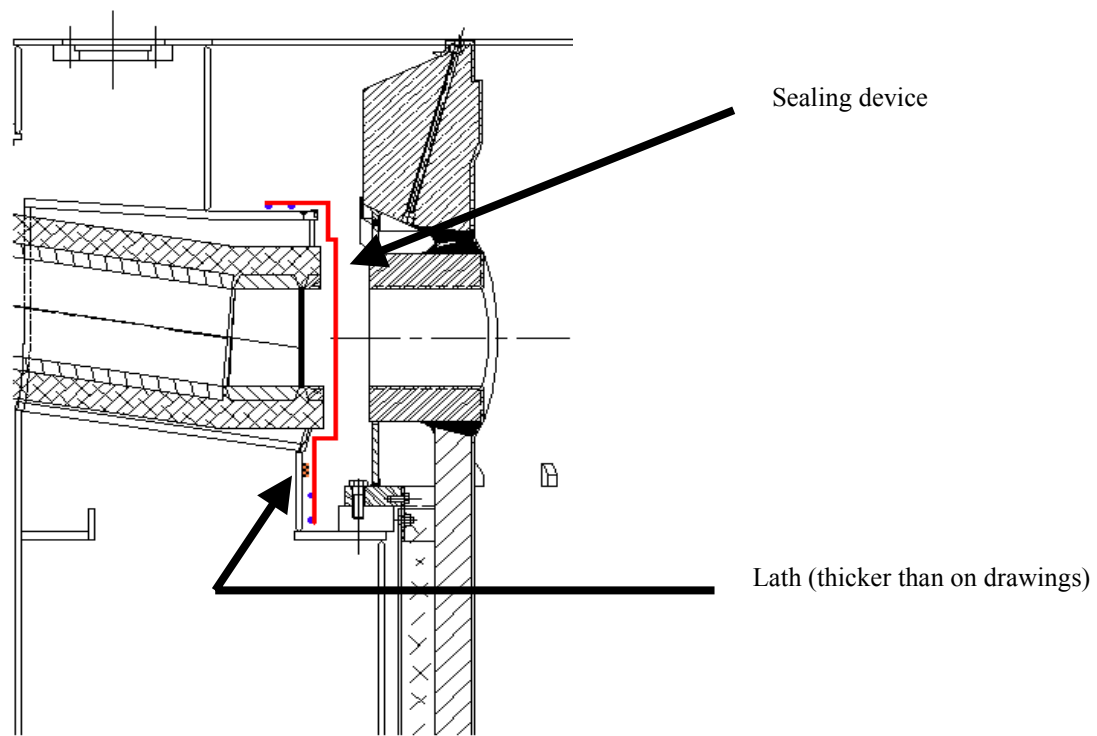


Figure 7. First major problem: the lath (thicker than foreseen on drawings) makes the positioning of the sealing device impossible.

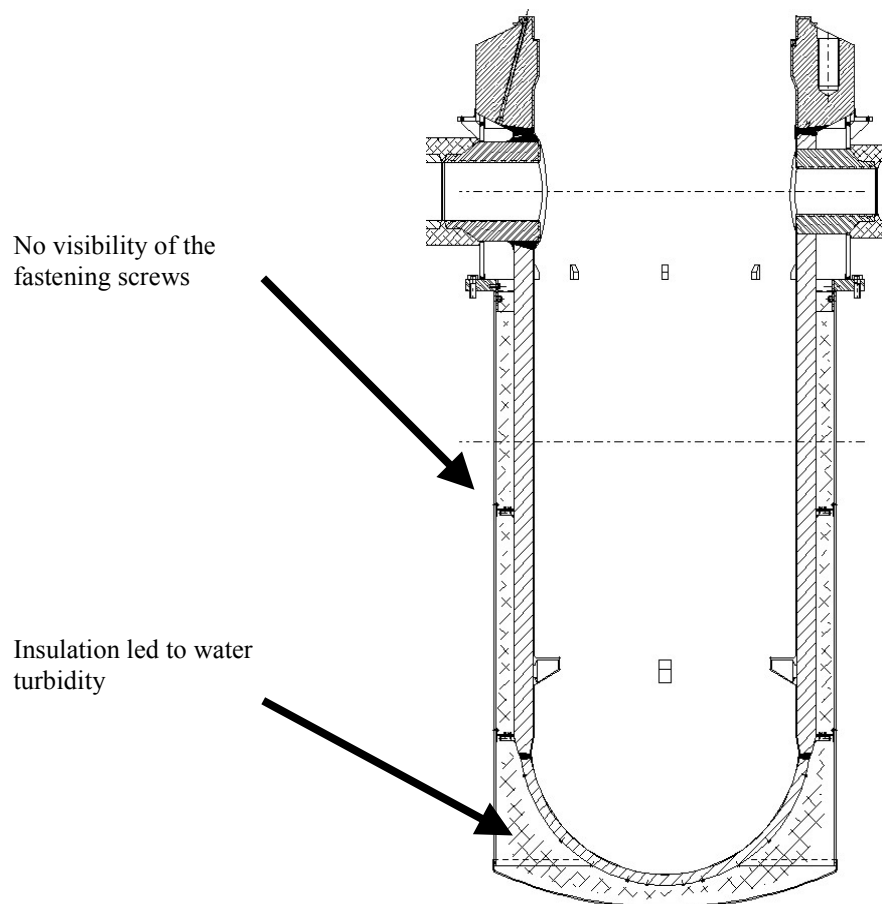


Figure 8. Second major problem: no visibility of the screw implied additional operation and the insulation led to water turbidity.

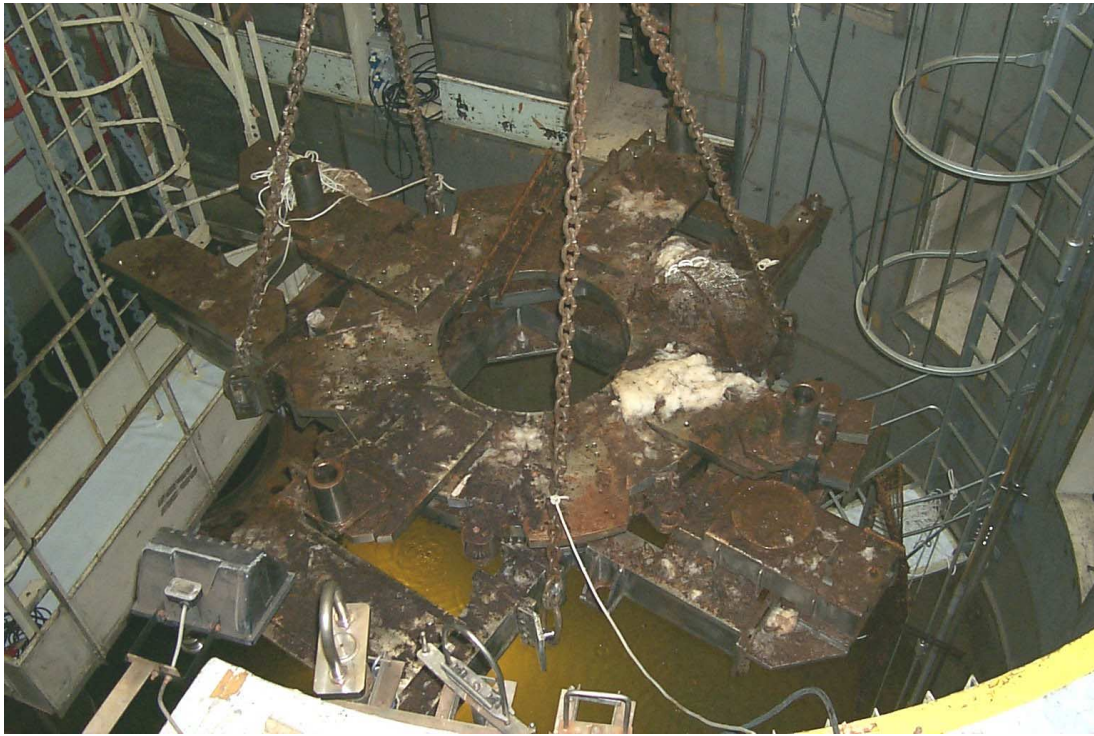


Figure 9. View of the turntable removed from the pool after the turbidity problem.

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Summary of feasibility studies on in situ disposal as a decommissioning option for nuclear facilities

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Abstract. A scoping study was conducted over the period 1998–2000 to consider the feasibility of in situ disposal as a decommissioning option for AECL’s Nuclear Power Demonstration Reactor located at Rolphton, Ontario. The results of a detailed assessment are summarized and the study concludes that in situ disposal appears feasible. Additional work required to confirm the results is also identified. A second in situ component, contaminated Winnipeg River sediments at AECL’s Whiteshell Laboratory located in Manitoba, was also evaluated. That study concluded that in situ abandonment would have no adverse impact on aquatic life, humans and the environment. A summary of the study is presented as an appendix to the report.

1. Introduction

Studies have recently been conducted by AECL to consider *in situ* disposal or abandonment as a decommissioning option for individual reactor facilities and for components of supporting systems and infrastructure. The results of two studies are summarized in this report.

The first project was a scoping study undertaken in 1998/99 to assess the feasibility of *in situ* disposal of the NPD reactor (Figure 1) near Rolphton, Ontario. The results are detailed in “Scoping Assessment of the In situ Disposal of the Nuclear Power Demonstration (NPD) Reactor”[1]. A brief summary of the study and the documented results are presented in the main body of this report.

The second study addresses contaminated river bottom sediments at the process water outlet from AECL’s Whiteshell Laboratory. This was the primary release point for cooling water from the Whiteshell Reactor, WR-1. The results of this study are detailed as part of the environment assessment for the Whiteshell Decommissioning Project [2]. A brief summary of the operation, approach and results are presented in Appendix 1.

2. The NDP feasibility study

2.1. Background and objectives

The NPD reactor was a heavy water moderated and cooled pressure tube reactor which was shutdown in 1987 after 25 years of operation. All nuclear fuel and operating fluids were removed from the facility as part of an initial decommissioning phase completed in the early 90’s. Most of the other radioactive components remain inside the reactor building. Both activation products and fission products are present.

In situ disposal involves remedial action to limit the mobility and release of the radioactive contamination such that NPD will be in a passively safe state. In particular, the present concept of *in situ* disposal consists of the following activities:

- Removing all uncontaminated surface structures
- Dismantling the upper contaminated structures such as the primary heat transport circuit and placing these structures at a lower level within the reactor building
- Filling all the voids in the reactor building with a mixture of swelling clay and sand
- Placing an impermeable clay-based cover over the top of the reactor building within the overburden layer
- Plugging the drain path from the reactor building to the pump house with impermeable material
- Removing more highly active reactor components if necessary

In this report, a scoping-level assessment is made of the radiological consequence of this *in situ* disposal. The radionuclides addressed here are those found in previous assessments of disposal of L&ILW to have the largest radiological consequence and those with large inventories at NPD. A more detailed assessment may require consideration of some other radionuclides.

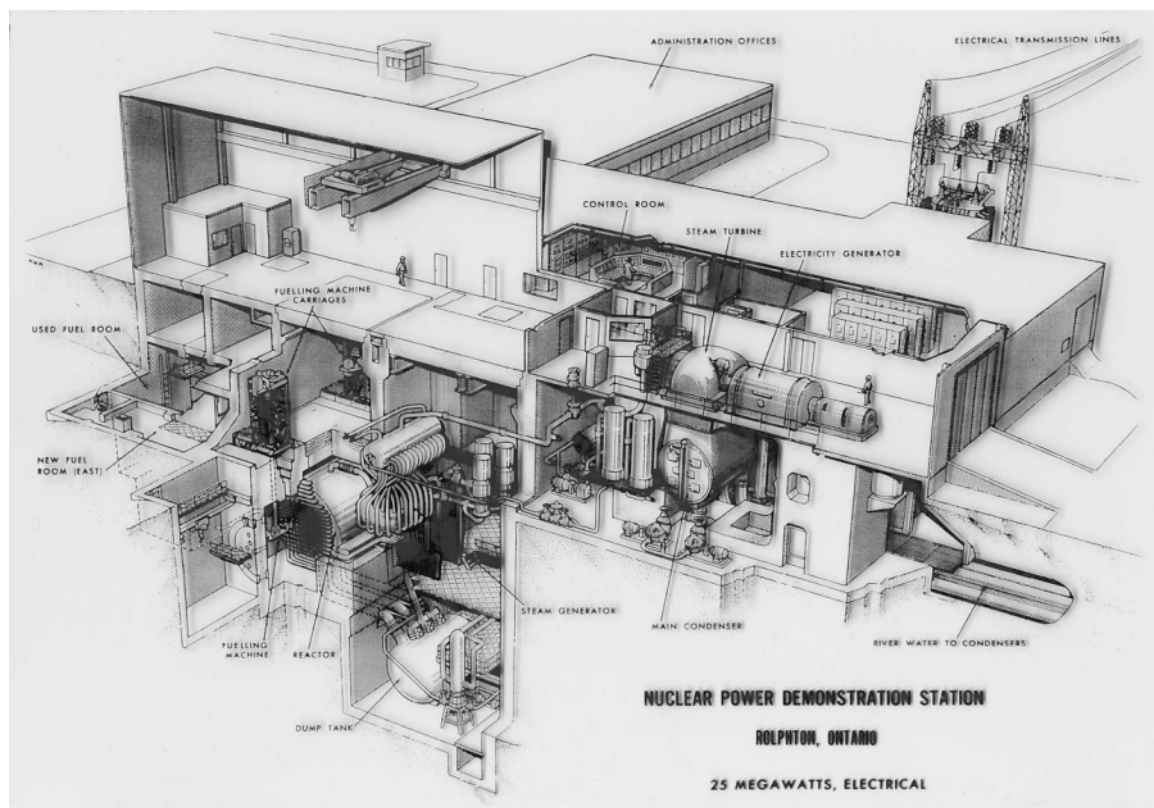


Fig. 1. Three dimensional drawing of NPD.

3. Scoping assessments modeling

The scoping study strives to assess the radiological consequences of in situ disposal of the reactor. The radionuclides addressed cover the entire range of radionuclides identified through review of waste disposal records and inventory assessments. The radionuclide sources are summarized in three distinct categories: 1) aluminum core components, 2) concrete forming the biological shield, 3) carbon steel of the vault liner and tube end supports, 4) stainless steel end fittings and closure plugs, 5) zircaloy pressure tubes and 6) loose surface contamination. The key radionuclides associated with each source are given in Table 1.

Three assessment scenarios were examined through modeling as follows:

- Release into drinking water
- Leaching and migration into ground water followed by lifestyle exposure
- Direct human intrusion into the site

The key activity involved modeling of the disposal vault, the geosphere and biosphere to identify/assess pathways and release mechanisms. The impact of the release paths on key receptors was evaluated for each model. Summary information on modeling for each component is summarized below.

The Near-Field (Vault) Model

In this study, the interior of the reactor building, the walls of the reactor building, the cement backfill around the building and the blast-damaged rock from the reactor excavation are collectively referred to as the ‘vault’. The model developed simulates the release of radionuclides from their sources and their subsequent transport through the water filled pore spaces in the reactor, through the walls of the reactor building and through the cement backfill and blast-damaged rock into the surrounding bedrock. The model includes the following processes:

- Release of radionuclides
- Two dimensional transport by advection and dispersion
- Degradation of engineered barriers
- Linear sorption onto solids
- Spatial variability in the vault,
- Time-dependent solubility limitations on transport
- Gas-generation
- Radioactive decay and in-growth

The Geosphere Model

The objective of the geosphere model is to evaluate the mass transport rate of radionuclides released from a disposal facility as a function of time, the transport path and the discharge zone, including the possible effects of a groundwater withdrawal well.

The model was developed in several stages:

- A conceptual model was developed based on expert judgment and the known local and regional geological and hydrogeological conditions.

- Data used were selected or collected.
- Detailed two dimensional and three dimensional groundwater flow models were developed to help identify the dominant pathway(s) for the transport of water from the vault to the discharge zone and the characteristics of the discharge zone. The groundwater flow along the transport pathway was characterized by reference hydraulic heads or groundwater flow velocities.
- For possible implementation in a probabilistic performance assessment framework, a network transport model was developed to reflect the previously identified dominant flow path(s) from the vault to the discharge zone(s).
- Mass transport calculations for radionuclide decay chains were done along the pathways of the network representation of the groundwater flow field to give rates of discharges to the biosphere.

Table 1. Estimated NPD radionuclide inventories for various waste sources (in bq at 50 years after plant shutdown)

Radio-nuclide	<i>AECB Scheduled Quantities⁽¹⁾</i>	ALUM ⁽²⁾	HDCE ⁽²⁾	ILWI ⁽²⁾	SCAR ⁽²⁾	STAN ⁽²⁾	ZIRC ⁽²⁾
¹⁴ C	3.7x10 ⁶	5.25 x 10 ¹¹	1.34 x 10 ⁶	7.9x10 ⁷	3.26 x 10 ⁴	8.49 x 10 ⁵	1.13 x 10 ¹²
³⁶ Cl	3.7x10 ⁵	1.45 x 10 ¹⁰	2.2 x 10 ⁹	3.0x10 ⁵	6.96 x 10 ⁵	7.12 x 10 ⁶	2.53 x 10 ¹⁰
⁶⁰ Co	3.7x10 ⁶	2.14 x 10 ¹¹	5.08 x 10 ⁶	n/a	2.02 x 10 ⁹	1.04 x 10 ¹¹	4.80 x 10 ¹¹
¹³⁵ Cs	Not specified	n/a	n/a	3.6x10 ³	n/a	n/a	n/a
¹³⁷ Cs	3.7x10 ⁵	n/a	n/a	1.5x10 ⁹	n/a	n/a	n/a
⁵⁵ Fe	3.7x10 ⁶	n/a	n/a	n/a	1.50 x 10 ⁷	8.29 x 10 ⁸	2.83 x 10 ⁸
¹²⁹ I	Not specified	n/a	n/a	7.0x10 ⁵	n/a	n/a	n/a
⁵⁹ Ni	Not specified	n/a	n/a	n/a	n/a	1.86 x 10 ⁹	1.71 x 10 ¹¹
⁶³ Ni	3.7x10 ⁵	n/a	n/a	n/a	n/a	1.70 x 10 ¹¹	2.31 x 10 ¹³
⁹⁰ Sr	3.7x10 ³	n/a	n/a	9.6x10 ⁸	n/a	n/a	n/a
⁹⁹ Tc	3.7x10 ⁵	n/a	n/a	3.7x10 ⁵	n/a	n/a	n/a

⁽¹⁾ Atomic Energy Control Board (renamed the Canadian Nuclear Safety Commission in 2000) Scheduled Quantities. These are quantities of radionuclides specified as exempt from licensing (AECB 1978, Sections 3 and 6, and Schedule I). These quantities are presented for perspective on the quantities in the vault inventories. As can be seen, the inventories of radionuclides in NPD are many orders of magnitude greater than the Scheduled Quantities.

⁽²⁾ Radionuclide sources considered

ALUM Aluminum metal components – calandria tubes, calandria shell, end reflector stepped-tube

HDCE High density concrete forming the biological shield of the reactor

ILWI Instantly released surface contamination

SCAR Carbon steel components – tube end supports, steel vault liner

STAN Stainless steel-end fittings, closure plug assemblies, fuel latch assemblies, fuel spacer sleeves

ZIRC Zircaloy coolant pressure tubes

n/a – not applicable

The Biosphere Model

The biosphere model estimates the ultimate fate of any radionuclides that escape the vault and the geosphere. In the biosphere, the radionuclides are partitioned among solids such as sediments and soils, liquids in surface water bodies, and gases and suspended particles in the atmosphere. The radionuclides are also partitioned to biota, both human and non-human. The

estimated dose rate to an individual of a critical group of humans was estimated. Essentially this is a group of individuals who live in the worst place and time with regard to receiving a dose from emissions from the vault. In this assessment, the radiation dose consequences for non-human biota are not computed. The ultimate fate of all radionuclides in the biosphere model is,

- decay to stable nuclides,
- flushing downstream and
- dispersal into the atmosphere.

4. Results

The potential doses to humans are based on scenarios that describe the reasonable use of the contaminated biosphere resources. The critical group has been defined in a way that is conservative for dose estimation. Among other attributes, it gathers all its resources from the contaminated area. Exposure pathways considered are listed in Table 2.

The central scenario analyzed based on the modeling developed is leaching into groundwater followed by lifestyle exposure. The maximum dose rate to the most exposed individual of the critical group was determined for all of the radionuclides considered in this study. The maximum dose rate for the top ten contributors are given in Table 3 and the dose rate as a function of time, which is dominated by ^{36}Cl , is shown in Figure 2.

Human intrusion models were also assessed to evaluate the impact of direct contact with the in situ waste inventory. The most likely access considered was excavation associated with a construction scenario or the drilling of a water well into the storage vault. Such scenarios are highly dependent on probability of a direct intrusion. The scenario modeled, in the near vault area, was drilling of a well which intercepts the entire radionuclide plume from the vault area. Table 4 shows the maximum dose rates for that scenario. The highest maximum dose calculated is 2.2×10^{-5} Sv/a which is only a factor of two below the dose rate associated with the Canadian Nuclear Safety Commission (CNSC) risk limit for radioactive waste disposal of 5.0×10^{-5} Sv/a.

Several variant simulations were also analyzed to determine how remedial actions and/or variations in model assumptions might impact the results. Remedial actions considered were:

- Removal of the reactor;
- Removal of reactor components and portions of the biological shield;
- Isolation of the vault by use of sealing clay layers;
- Hydraulic isolation of NPD from ground water flow in the surrounding construction cavity;
- Institutional controls to prevent human activities around the site.

Other simulation variants considered ^{14}C solubility increases, effects of an alkaline plume in the reactor vault, re-flooding of NPD, errors in the radionuclide inventory and increased hydraulic conductivity of clay backfills.

A summary of the dose rate results of the variant simulations is presented in Figure 3. Maximum dose rates for each variant are compared to the maximum dose rate for the central scenario and to the dose rate associated with the CNSC risk limit in Figure 4.

There are many uncertainties associated with the modeling assumptions and data. The results of the analysis of variant simulations indicate that only the uncertainty in initial radionuclide inventory might lead to exceedance of the CNSC risk criteria.

The variant simulations analyzed show that the following remedial measures would be effective in reducing the radiological risk at NPD and increasing the margin of safety for *in situ* disposal.

- Removing reactor components and contaminated portions of the biological shield;
 - Maintaining institutional controls after disposal;
 - Adding a clay barrier for the unprotected portions of the biological shield;
- Isolating NPD from the local groundwater flow.

Table 2. Exposure pathways to humans considered in the biosphere model

	Pathway	Description
1. 2. 3. 4.	Uptake from Soil Soil/plant/meat/human Soil/plant/milk/human Soil/plant/bird/human Soil/plant/human	Used to estimate dose rate to a member of the critical group from the ingestion of plants and animal products that have been contaminated by uptake from soil.
5. 6. 7. 8.	Atmospheric Deposition Air/plant/meat/human Air/plant/milk/human Air/plant/bird/human Air/plant/human	Used to estimate dose rate to a member of the critical group from the ingestion of plants and animal products that have been contaminated by deposition on plant leaves.
9. 10. 11.	Animal Air Inhalation Air/meat/human Air/milk/human Air/bird/human	Used to estimate dose rate to a member of the critical group from the ingestion of animal products that have been contaminated by the air that the animals inhale.
12. 13. 14. 15. 16. 17. 18. 19.	Ingestion of Water Water/meat/human Water/milk/human Water/bird/human Water/human Ingestion of Soil Soil/meat/human Soil/milk/human Soil/bird/human Soil/human	Used to estimate dose rate to a member of the critical group from the direct ingestion of drinking water, and from the ingestion of animal products that have been contaminated by drinking water and by irrigation water. Used to estimate dose rate to a member of the critical group from the ingestion of contaminated soil; includes direct ingestion of contaminated soil (#19) and ingestion of animal products from animals that have ingested contaminated soil (#16–18)
20. 21.	Other Internal Routes Fish/human Inhalation	Used to estimate dose rate to the critical group from the ingestion of fish in a lake contaminated with radionuclides and from the inhalation of air contaminated with radionuclides.
22. 23. 24. 25. 26.	External Routes Air Water Ground Wood Inorganic	Used to estimate external dose rate to the critical group from immersion in contaminated air and water, exposure to contaminated ground (groundshine), and exposure to buildings constructed of wood and inorganic materials.

Table 3. Maximum dose rate of the top 10 contributors for the central scenario

Radionuclide species	Maximum dose rate (Sv/a)	Time of maximum (a)
Cl-36 HDCE	1.24×10^{-5}	1.48×10^1
C-14 ALUM	2.72×10^{-6}	5.60×10^3
Cl-36 ALUM	1.14×10^{-6}	4.24×10^3
Sr-90 ILWI	1.39×10^{-7}	1.00×10^2
C-14 ZIRC	2.15×10^{-8}	1.00×10^4
I-129 ILWI	1.23×10^{-8}	1.00×10^3
Cl-36 ZIRC	6.28×10^{-9}	1.00×10^4
Tc-99 ILWI	1.86×10^{-9}	1.80×10^3
Ni-59 ZIRC	1.07×10^{-9}	1.97×10^4
C-14 ILWI	2.06×10^{-10}	1.59×10^2

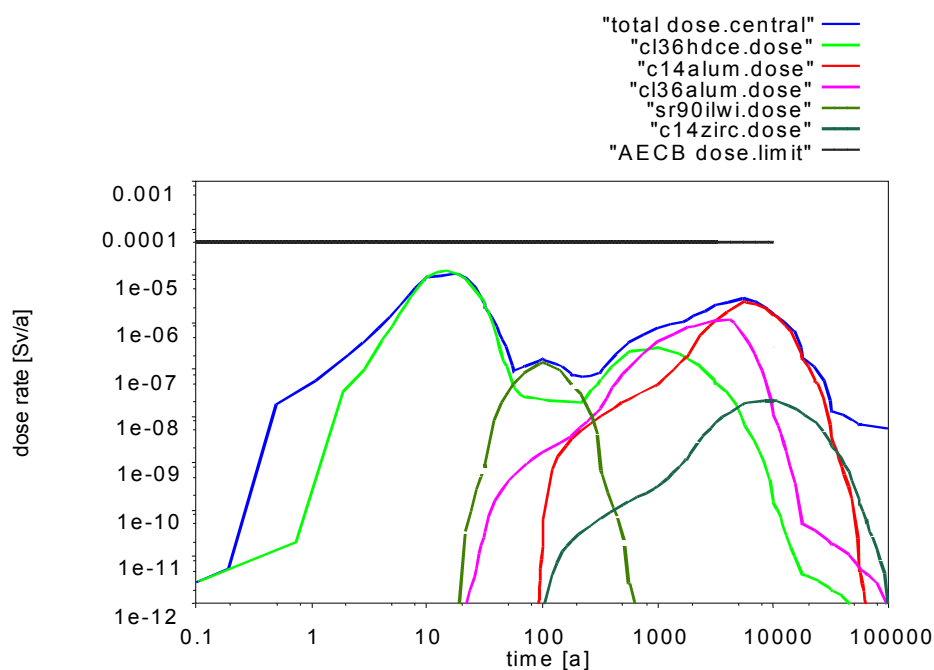


Fig. 2. Dose rate as a function of time for the central scenario.

Table 4. Maximum dose rates for a well adjacent to NPD

Radionuclide	Maximum near-field release rate (mol/a)	Time of maximum release rate (a)	Concentration (mol/m ³)	Dose conversion factor (Sv/Bq)	Maximum dose rate estimate (Sv/a)
C-14 ALUM	9.4×10^{-6}	6.7×10^3	6.3×10^{-9}	5.00×10^{-10}	4.6×10^{-6}
Cl-36 HDCE	1.1×10^{-3}	1.0×10^1	7.7×10^{-7}	1.00×10^{-9}	2.2×10^{-5}
Co-60 ALUM	5.8×10^{-15}	2.5×10^1	3.9×10^{-18}	2.86×10^{-9}	1.8×10^{-11}
Cs-135 ILWI	8.7×10^{-12}	7.9×10^1	5.8×10^{-15}	2.00×10^{-9}	4.3×10^{-14}
Cs-137 ILWI	1.1×10^{-11}	5.6×10^1	7.3×10^{-15}	2.00×10^{-8}	4.1×10^{-8}
Fe-55 SCAR	3.8×10^{-20}	1.8×10^1	2.5×10^{-23}	2.00×10^{-10}	1.6×10^{-17}
I-129 ILWI	4.6×10^{-7}	8.2×10^2	3.1×10^{-10}	n/a	7.8×10^{-7} *
Ni-59 ZIRC	5.7×10^{-7}	2.3×10^4	3.8×10^{-10}	6.70×10^{-11}	2.9×10^{-9}
Ni-63 ZIRC	1.2×10^{-11}	1.8×10^2	8.0×10^{-15}	2.00×10^{-10}	1.4×10^{-10}
Sr-90 ILWI	9.9×10^{-11}	2.1×10^1	6.6×10^{-14}	3.30×10^{-8}	6.3×10^{-7}
Tc-99 ILWI	1.8×10^{-9}	1.6×10^3	1.2×10^{-12}	6.70×10^{-10}	3.2×10^{-11}

* determined from a specific activity calculation based on the ratio of stable to radioactive iodine in the thyroid.

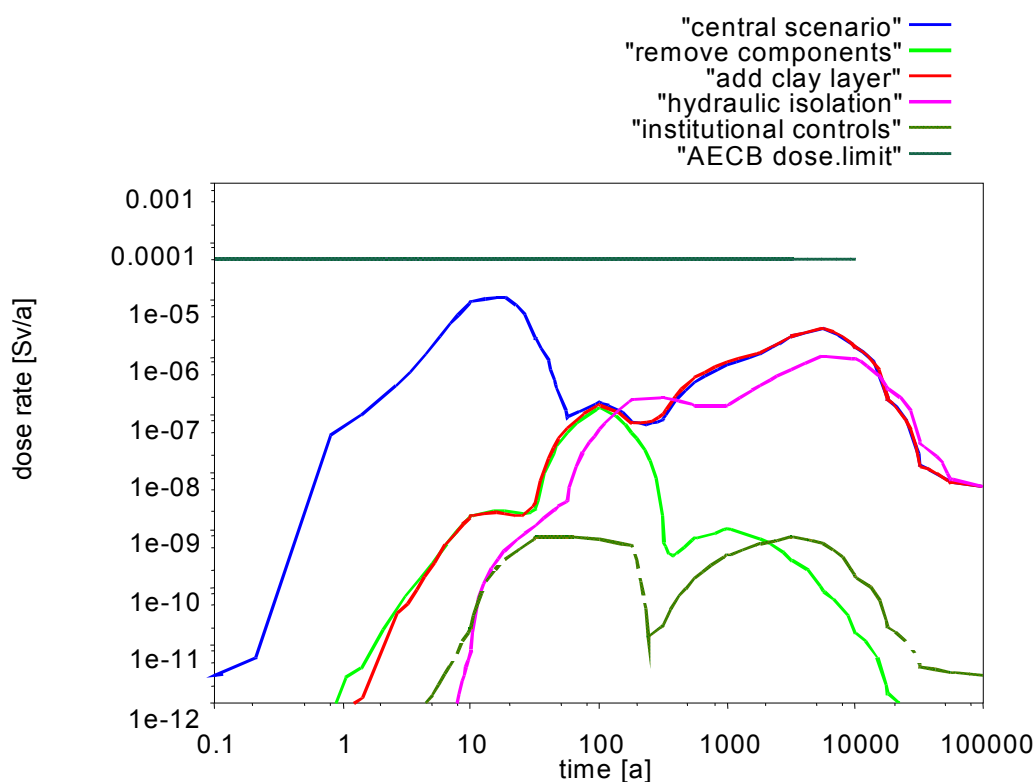


Fig. 3. Total dose rate following remedial measures.

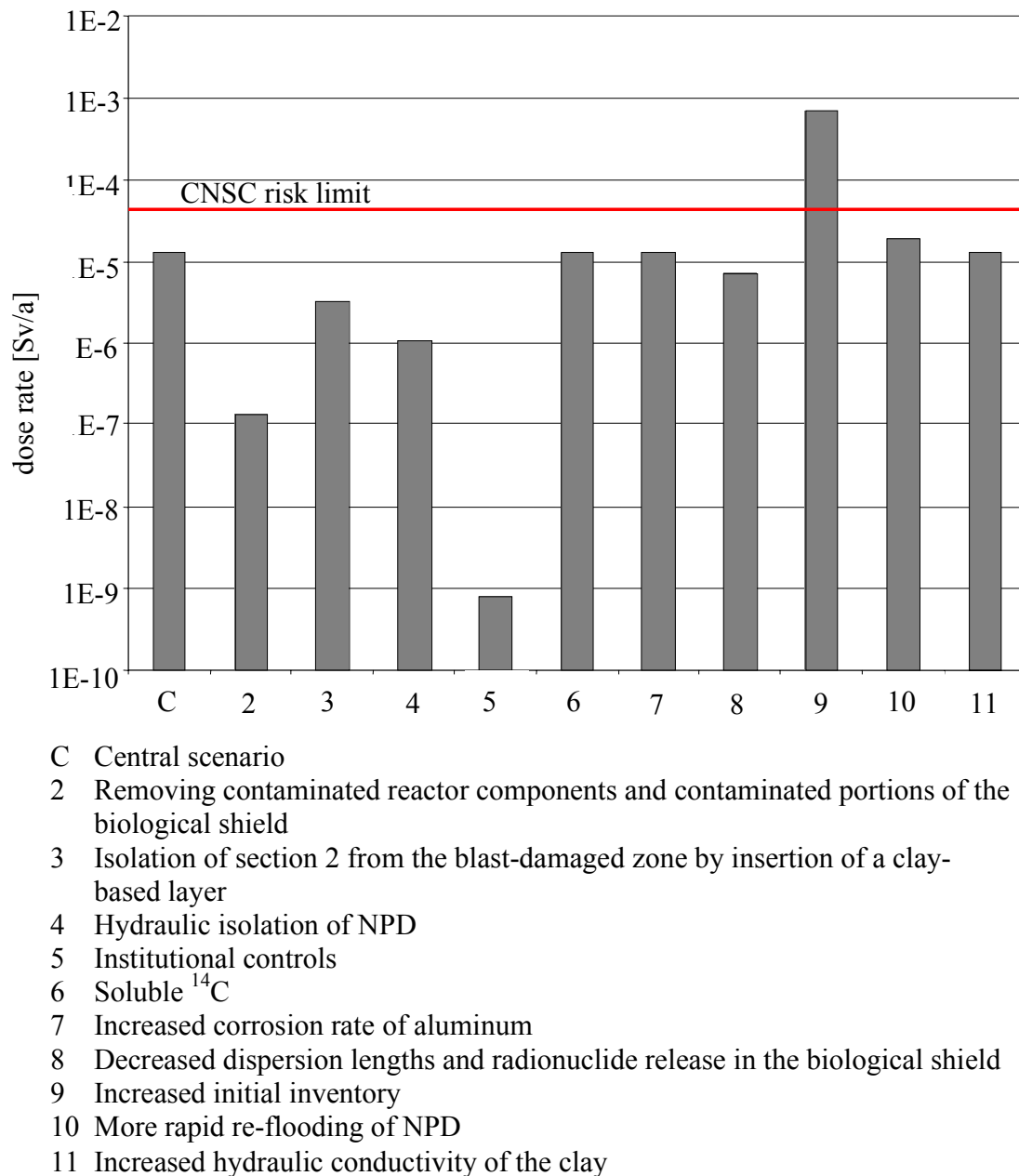


Fig 4. Maximum dose rates for variants compared to the central scenario and to the CNSC risk limit.

5. Conclusions/recommendations

Although the results of the NPD scoping analysis are not sufficient to conclude that in situ disposal is viable, in situ disposal cannot be ruled out as a feasible option. The results of the leaching, migration and lifestyle exposures indicate that they are below but very close to recommended public exposure limits. The results of remedial analysis show that successful application of any or all measures would enhance the margin of safety and improve the likelihood of securing regulatory approval.

Analysis of the human intrusion scenario (the most limiting scenario) indicated that in situ disposal is viable. From a dose impact perspective it is clear that application of institutional controls (see Figure 4) following disposal would enhance the safety of the human intrusion scenario.

A number of recommendations are advanced to indicate further analysis required to narrow the risk in implementing the in situ disposal option. These additional studies are considered necessary prior to advancing a proposal on in situ disposal of the NPD reactor. Specific recommendations for additional work are:

- Further studies and analysis be undertaken to narrow the uncertainty in radionuclide initial inventory at NPD, particularly the initial inventory of ^{36}Cl in the biological shield;
- A detailed groundwater flow and remedial action study should be conducted to determine and evaluate measures to achieve hydraulic isolation of NPD;
- A detailed engineering and contaminant transport study be undertaken to evaluate the effect and feasibility of adding a clay-based barrier between portions of the biological shield and the blast-damaged zone around the reactor;
- Methods of preventing human habitation in the environs of NPD subsequent to disposal should be investigated;
- The results of this study should be supported by a further analysis using a reliable, three dimensional contaminant transport code capable of arbitrary geometry;
- A program should be initiated to investigate optimum methods for the use of clay-based fill at NPD. Resources should be devoted to investigating the influence of cement degradation on the fill and to develop methods to avoid detrimental effects.

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Appendix 1

Summary of the Feasibility Study on In situ Abandonment of Contaminated Winnipeg River Sediments at the Whiteshell Laboratories Process Water Discharge

1. Introduction

The process water discharge from Whiteshell Laboratories releases radioactive and other contaminants to the Winnipeg River. The contaminants are continuously monitored and with few exceptions the releases are below the relevant standards. In routine operation, contaminated water is held in a tank (in the Active Liquid Waste Treatment Center or ALWTC), sampled, and released only if the release will meet the regulatory criteria. There have been a few accidental releases, all duly reported.

The releases changed distinctly over the operational history of the Laboratory. Releases were highest prior to 1985, when the WR-1 reactor was operating and the radionuclide mixture was characteristic of an operating reactor (Table 1). After 1986, the releases decreased progressively to the present. As a result of the routine release of radioactively contaminated aqueous waste during the operating period, radionuclides are still detectable in river sediments at the outfall area.

2. Objectives and methodology

The objective of this evaluation is to estimate the potential effects of the contaminated sediment on biota in the river and on humans and to evaluate the feasibility of abandoning the contaminated sediment in situ. The assessment endpoints for aquatic biota are clams because:

- they are abundant in the contaminated area,
- they dwell in or on the sediment,
- they have relatively small home ranges and so are exposed to a small area of sediment,
- they live long enough to accumulate radionuclides over several years, and
- they are important prey for fish, otters and turtles.

There are no realistic assessment scenarios leading to a notable dose for humans. The assessment scenario considers external exposure from the sediment, as could result if the sediment were dredged (very improbable), the sediment was exposed as shoreline (improbable), or selected items from the sediment were collected as keepsakes (very improbable).

The work involved the following steps:

- Developing a conceptual model of the River bottom and the general nature of the sediments;
- Adjusting the model with information obtained from a series of diver inspections of the River;
- Defining a survey area based on areas delimited by identifying criteria where there would be no effects on human or ecological health;
- Carrying out a gamma survey of the River bottom;
- Analyzing sediments;
- Analyzing clams (as an indicator of ecological risk); and
- Preparing dose estimates for clams and humans.

3. Defining the investigation area

A practical method to specify a technical cutoff value for a survey of the process discharge area was developed based on Environment Canada and Health Canada priority substance data [1]. This data presents the effects of radiation on organisms in increments of radiation level above background. A conservative level of 350 times background and a hyper-conservative level of 35 times background are proposed. for the purpose of the Winnipeg River sediments study an even more conservative cutoff level of 10 times background was selected to delineate the investigation area.

4. Sampling and data collection

Underwater divers were used to carryout a radiation survey to confirm the size of the assessment area. A gamma survey probe adapted for underwater use was carried by the divers. For a reading, it was pressed onto the sediment surface and counts were recorded by an operator on the surface. Readings were taken on a grid pattern downstream of the pipe. The positioning was determined by GPS.

In addition to the gamma survey probe, a 256-channel gamma spectroscopy probe was adapted for underwater use and was used to calibrate the survey probe from counts per second (cps) to nGy/h in several locations. Analyses of the top 5 cm of sediment were used for the calibration, because that approximates the depth of sediment ‘seen’ by the gamma probe. Once the center-of-plume was defined, three long cores were collected in split barrel cylinders and a composite of ten grab samples of surface sediment were collected. Three deep cores were also collected upstream of the pipe in an area considered to be representative of the sediment type at the center-of-plume. Composite samples were also collected in the same manner upstream of the pipe and in the downstream bay. Clams were collected as available.

4.1. Results and interpretation of gamma survey

The positions of the gamma survey points are shown in Figure A1. The points are colour coded to show the level of activity observed. These are also shown in the 3-D plot of Figure A.2. The plane at the top of Figure A2 is 350-fold above background, the conservative level referred to in section 3 above. Clearly, none of the observations approach this level. The rapid decrease in concentration with distance is evident.

The two peaks of activity just downstream of the pipe outlet were taken as the center-of-plume. The fact that there are two peaks instead of one may reflect features of the bottom topography or sediment (that were not apparent to the divers) or may reflect the discharge history. When discharges occurred at times of high flow, such as spring melt or when the reactor was drawing large volumes of water, the increased exit velocity of the effluent would propel contaminants further towards the center of the river. In low flow, the contaminants would move more directly downstream from the end of the pipe.

The gamma survey results were plotted with software that allows definition of isopleths and computes the area in square meters bounded by each isopleth line. These were used to estimate the inventory of contaminant in the sediment of the evaluation area (Table A1).

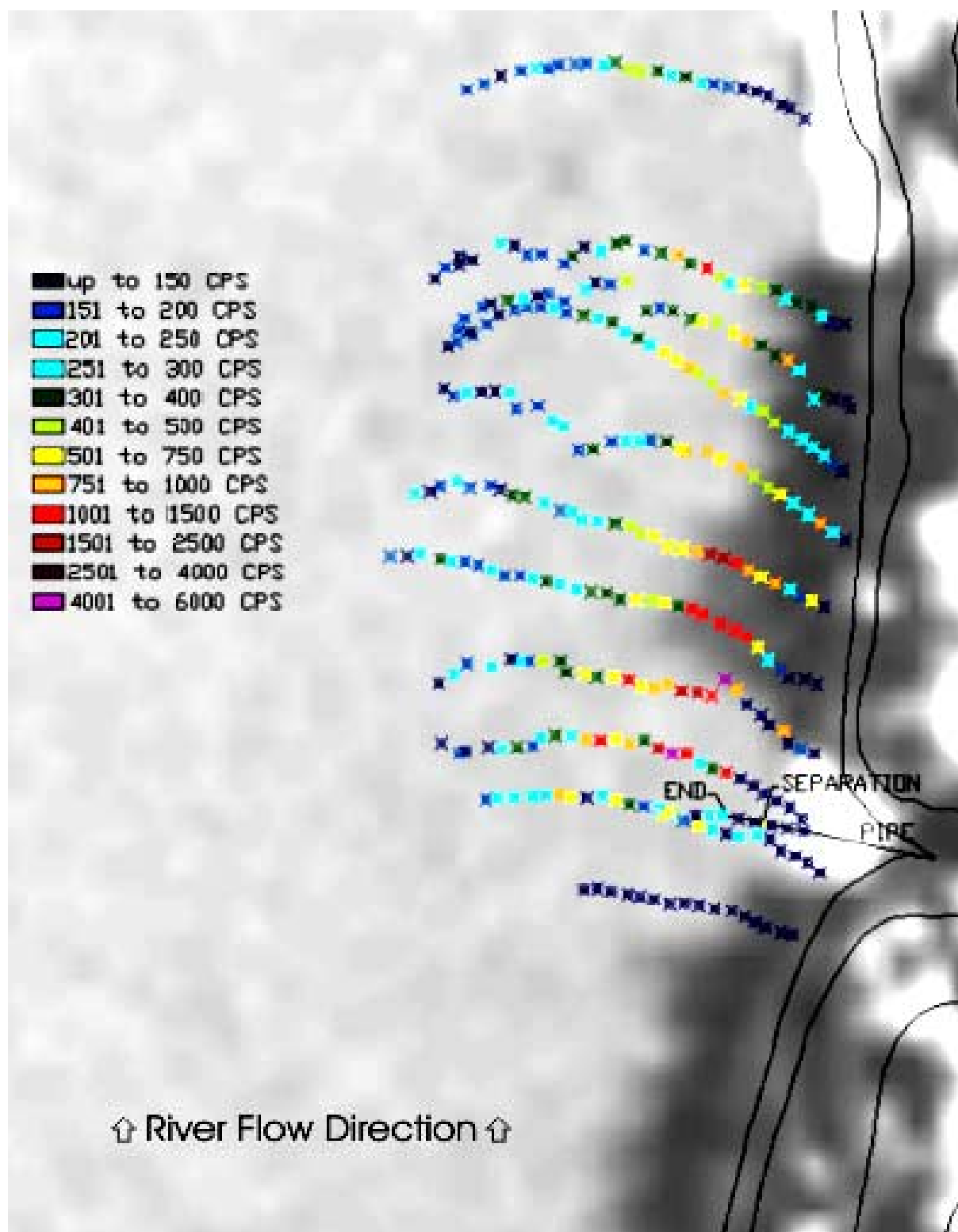
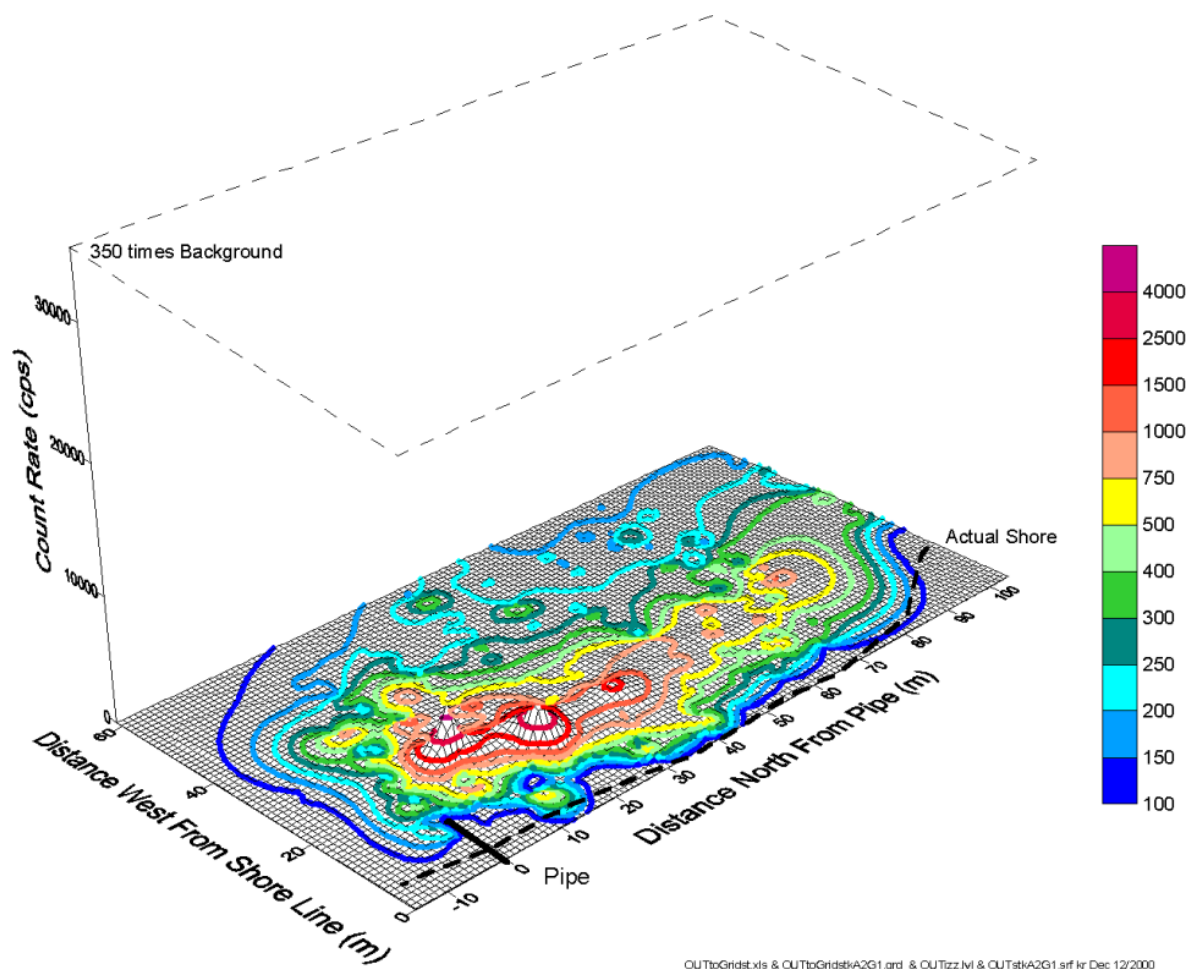


Fig.A1. Positioning of grid points for gamma survey of sediments.



COUNT RATE FROM OUTFALL STUDY AREA

Fig.A2. Three dimensional rendition of the gamma survey results.

The observed activity at the isopleth line (counts per second or cps) was corrected for background, converted to dose rate (nGy/h) using the calibration developed for the sediment survey and converted to concentration of ^{137}Cs (Bq/g) based on a calibration point where sediment analysis was completed for sediment from a gamma survey position. Assuming a sediment density of 1500 kg/m^3 , consistent with the dense clay observed and a contamination depth of 5 cm (discussed in detail in the next section), the contamination per unit area was computed. This value multiplied by the corresponding area between this isopleth and the next gives the inventory between adjacent isopleths in Bq. These values were summed for all the isopleths, resulting in an estimate of total inventory of 1.3 GBq. It is relevant to note that this is substantially less than the annual releases of ^{137}Cs prior to 1985 when the reactor was operating. Because there is no evidence of buried contamination below 5 cm, it is assumed that the ^{137}Cs absent in the local sediment was flushed downstream.

5. Evaluation

Clams were chosen as the assessment endpoint. The dosimetry calculations based on tissue concentrations are the same for all organisms, so only the potential for biomagnification

would indicate the need to evaluate higher trophic levels. Exposures to predators of the clams were not estimated because:

- The possible predictors are quite mobile and will feed outside the contaminated area, thus diluting their ingestion of contamination by some unknown amount; and
- Most radionuclides do not biomagnify, and are at their highest concentration in biota most closely associated with the contaminated media.

Human exposures from ingestion from the sediment are not likely. Clams and other benthic invertebrates such as crawfish are not commonly consumed by people in this area. Fish may be contaminated, but monitoring of fish has shown only slightly elevated contamination downstream. As a result, only external exposure is plausibly important and even this is improbable. The scenario chosen was external exposure from proximity to the sediment. This is assumed to include scenarios such as handling a boat anchor that was rooted in the sediment.

Table A1. Estimation of the inventory of ¹³⁷Cs in sediments in the study area just downstream of the outfall

Isopleth (cps)	Net Count Rate (ncps)	Isopleth Area (m ²)	Net Area (m ²)	Isopleth Dose Rate (nGy/hr)	Activity Concentration (Bq/g)	Total Activity (GBq)	Background (cps)
100	0	1598.71	6.8	8	0.27	0.000	100
150	50	1591.90	3.9	23	0.81	0.000	
200	100	1587.99	15.6	38	1.34	0.002	
250	150	1572.39	44.3	54	1.88	0.006	
300	200	1528.14	50.1	76	2.69	0.010	
400	300	1478.03	193.6	107	3.76	0.055	
500	400	1284.44	244.9	161	5.65	0.104	
750	650	1039.54	556.1	237	8.34	0.348	
1000	900	483.45	204.2	352	12.37	0.189	
1500	1400	279.21	159.3	581	20.44	0.244	
2500	2400	119.92	89.1	963	33.88	0.226	
4000	3900	30.81	28.7	1346	47.33	0.102	
5000	4900	2.15	2.1	1560	54.86	0.009	
		Total=	1598.706			Total =	1.295

The estimated internal dose to clams (Gy/a) is the product of the internal DCF (Dose Conversion Factor) ((Gy/a)/(Bq/kg fresh tissue) and the estimated tissue concentration (Bq/ kg fresh tissue). The estimated external dose to clams is from sediment immersion only (because it has a 1500-fold greater DCF than from water immersion and the K_d values are all large), and is the product of the sediment concentration and the external DCF divided by the sediment wet/dry weight ratio:

$$(\text{Bq/kg dry sediment} * (\text{Gy/a})/(\text{Bq/kg wet sediment})) / (\text{kg wet sediment/kg dry sediment}) = \text{Gy/a}$$

Total dose from each radionuclide is the sum of the internal and external dose estimates. The total dose for clams in the composite sediment was 0.017 mGy/a, well below even the most conservative dose guideline[2] of 50 mGy/a. For the 99.9th percentile case, the total estimated

dose was 6.7 mGy/a, still below the guidelines. This indicates there is very little potential for harm to populations of clams, and the analysis is sufficiently conservative that this can be extended with some confidence to all organisms living in or on the sediment or feeding from the sediment.

Dose to humans in proximity to a semi-infinite plane of contaminated sediment for 1% of the year was computed. The concentrations of the 99.9th percentile case were used, because these concentrations were higher than any observed. The total dose rate for this very conservative case was 0.04 mSv/a, below the risk-based criteria of 0.05 mSv/a. Any actual risk would be many orders of magnitude lower because:

- 1% occupancy is nearly impossible along this shoreline or in other exposure scenarios;
- The total spatial extent of contaminated sediment in the investigation area is small and does not constitute a semi-infinite plane; and
- There was no sediment found to have the 99.9th percentile concentration and if it exists it will be a very small volume of sediment.

To summarize the dose estimates, there is a very low probability of harm to non-human biota or humans from the sediment contamination left in the present location. With engineered or natural displacement, the potential for impact is even lower because of further dispersion and dilution in the river. In effect, the operation of the Whiteshell Laboratories within its regulated release permits has led to no significant impact in the river sediments, a confirmation that the original planning was sufficiently well founded.

6. Conclusions

The conclusions of this evaluation deal with the description of the contamination present in the sediments, the outfall itself, and the possible doses to non-human biota and humans. In point form:

- The center-of-plume is downstream and outward from the pipe outlet.
- There is a rapid decrease in sediment contaminant concentration with distance from the outfall.
- There are very localized spots of higher activity,
- Only a very small fraction of the radionuclides released is still present in the sediment near the outfall.
- Even with extremely conservative dose estimation methods, the doses to non-human biota (clam as the specific endpoint) and humans (based on external exposure) are below accepted guidelines.

The abandonment of the contaminated sediment in situ is considered feasible as the final endstate for the process water outfall area.

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The DR-2 decommissioning project, Denmark

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Abstract. The DR-2 reactor of the Risoe National Laboratory was closed down in 1975, the fuel removed, the circuits drained and the reactor sealed. In 1997 the DR-2 Study Project was initiated to determine the remaining radioactivity in the reactor and to plan the final decommissioning. So far all movable components have been removed from the reactor tank, measured and stored. The same is true, with two exceptions, for the hold-up tank room and work is under way on the components of the igloo at the thermal column. Later the thermal column, beam tubes and the interior of the primary circuit will be examined and holes will be drilled through the concrete shield. The lessons learned during the project are discussed.

1. Introduction

During the early years of the Risoe National Laboratory, i.e. around 1960, a number of nuclear facilities were built. They included three research reactors: the DR-1, a 2 kW homogeneous reactor, the DR-2, a 5 MW tank-type reactor, and the DR-3, a 10 MW heavy water moderated reactor with highly enriched fuel.

The first of these reactors to be closed down was the DR-2. It ceased to operate in 1975 since experience had shown that the DR-3 could cover all Danish needs for neutron beam experiments and for neutron irradiations. As there was some discussion about the possible future need for the DR-2, it was decided to close the DR-2 down in such a way that it could be restarted fairly easily. However, after a few years it became clear that there would be no need for the DR-2, and it was decided in the late seventies that the DR-2 would never be made operational again.

Since Denmark has no repository for low and medium radioactive waste and since the DR-3 continued operating it was decided to not to start the dismantling of the DR-2 immediately, but to wait until the DR-3 reached the decommissioning stage. Therefore the actions undertaken at DR-2 were limited.

Just after the close down in 1975 the fuel was removed from the DR-2 and sent to the United States. Further, the water was drained from the reactor circuits to reduce corrosion, and the reactor top was provided with a 5 cm lead layer to reduce the radiation from the reactor internals. In the late seventies the secondary circuit was dismantled, the reactor top was provided with additional shielding, a total of 10 cm of lead and a 40 cm thick concrete lid, and the reactor block was sealed. After these modifications the reactor hall was used until around 1996 for chemical engineering experiments.

In October 1997 a study project for the decommissioning of the DR-2 was initiated, and the aim of this paper is to discuss this project. However, it should be mentioned that just before Christmas 1999 a leak developed in the DR-3 reactor tank. After the leak had been closed DR-3 was started up again, but it soon seemed as if there was yet another leak. It was

therefore decided to perform a major examination of the reactor tank. The detailed examination of the DR-3 tank concluded that there was no second leak, but it raised doubt about the long-term integrity of the tank. Therefore the Risoe management decided in September 2000 that all nuclear facilities at Risoe should be closed down and decommissioned. This development will not be considered here, but it has had considerable influence on the DR-2 Study Project. The reason is that it is personnel from the DR-3 that carry out the work on the DR-2 Project, and work on the DR-3 was for obvious reasons given first priority.

2. Project objectives

The objectives of the DR-2 Study Project are to:

- investigate the state of the reactor;
- determine where which radionuclides remain in the DR-2, and in what amounts;
- plan the final decommissioning of the DR-2; and
- use the experience gained during this project to assist similar projects in other countries.

The work performed so far has concentrated on investigations on the state of the DR-2 and determination of the remaining radionuclides in the reactor. Further, a co-operation project on decommissioning of research reactors has been initiated with the Salaspils reactor in Latvia.

The planning of the final decommissioning of the DR-2 is, considering the decision to close down all reactors at Risoe, likely to become part of the overall Risoe decommissioning programme.

3. The DR-2

The DR-2 was, as mentioned above, a 5 MW tank-type research reactor with MTR-type fuel elements containing highly enriched uranium. The moderator and coolant are ordinary water. A vertical cross section of the reactor is shown in Figure 1.

The fuel elements were placed in a grid plate with 48 positions for fuel and reflector elements in a 6×8 array. Five shim-safety rods and one stainless steel regulating rod for fine control controlled the reactor. The shim-safety rods were flattened aluminium tubes containing B₄C powder and two lead-filled stainless steel tubes to increase the insertion rate. All control rods moved in special fuel elements containing a water gap for the rods in the middle and only half the number of fuel plates as compared to the standard elements. Stainless steel guide tubes, which guided the control rods down into the core, were mounted on top of the special fuel elements. The control rod drive mechanisms were located at the top of the reactor (not seen in Figure 1).

Initially the DR-2 used reflector elements of graphite, clad in aluminium. However, they were replaced by beryllium reflector elements in the early sixties because of leaks. Usually 12 reflector elements were used.

Six instrument tubes with neutron detectors (ion chambers) for use in the control of the chain reaction were situated below the core and the grid plate. Gamma radiation from the core

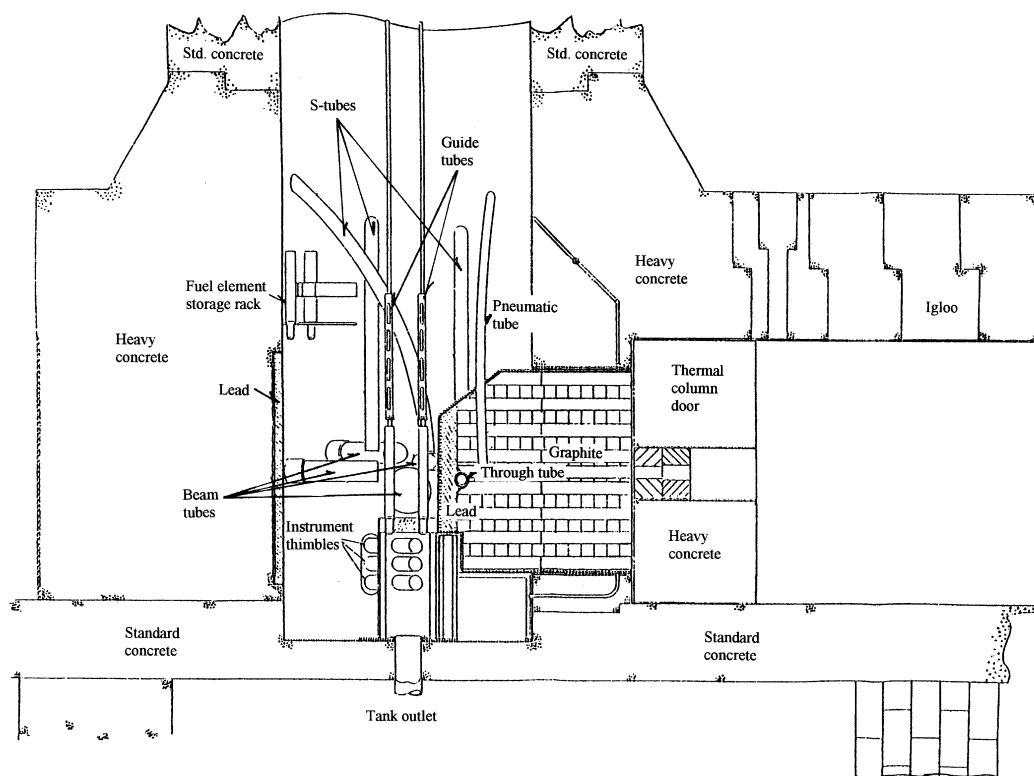


Fig. 1. Vertical cross section of the DR 2.

interfered with the proper functioning of the ion chambers, so a lead shield (not shown in Figure 1) was placed above and around the instrument thimbles.

The DR-2 is provided with a graphite thermal column, eight beam tubes, one through tube penetrating the thermal column, six bent S-tubes used for sample irradiation and two pneumatic tube systems. Two of the S-tubes and the pneumatic tubes penetrated down into the thermal column. A so-called V-tube facility consisting of five vertical tubes, intended for high flux irradiation of small samples, replaced one fuel element in the core. Also, there were vertical tubes in some of the beryllium reflector elements for the same purpose. Four of these were provided with 2" and four with 3/4" diameter vertical holes. At the inner end of the thermal column there is a lead shield to reduce the γ -radiation in the column. A movable heavy concrete door and a concrete "igloo" are located at the outer end of the column.

A storage rack for fuel elements, control rods, guide tubes etc. is situated against the reactor tank wall, some distance above the core.

Outside of the tank, the reactor is provided with a concrete shield. Between this shield and the tank wall is a lead thermal shield. The concrete outside the lower part of the reactor tank is heavy (barytes) concrete. The top part of the shield is made from ordinary concrete.

The coolant flow was downwards through the core to a hold-up tank situated in the room below the reactor tank. Here the ^{16}N -activity, produced by fast neutron induced $^{16}\text{O}(n,p)^{16}\text{N}$ reactions in the core, was allowed to decay before the coolant proceeded to the primary pumps and the heat exchangers in the basement of the DR-2 building.

A more detailed description of the DR-2 may be found in [1].

4. Organisation

The DR-2 is, according to the Danish regulatory authorities, “a reactor under decommissioning”. The head of the DR-3 is responsible for the DR-2, but he appoints one staff member to be the daily manager of the DR-2 and another to be the supervisor of the DR-2.

A planning committee, composed of two or three representatives from relevant Risoe sections and groups, did the planning for the project. The groups involved were:

- DR-3
- Reactor Safety Group
- Waste Treatment Plan
- Applied Health Physics Section

The reason for having more than one person from the same section or group was to have both an older, experienced person and one from the younger generation.

In addition, there was one representative from the Building and Construction Services and the former head of the DR-2 reactor, now working in the Safety Secretariat. This committee met initially once a month, but less frequently later on when the project planning was finished. The chairman of the committee was the leader of the DR-2 Study Project who come from the Reactor Safety Group.

When the actual work on the DR-2 was started, a steering committee, chaired by the project leader, carried out the short term planning and execution of the work. The other members of the steering committee came from the DR-3, the Applied Health Physics Section and the Safety Secretariat.

5. Achievements

When the DR-2 Study Project started in 1997 the reactor was at Stage 1 of the IAEA scale (see [2]). This meant that all fuel had been removed from the reactor site, all circuits had been drained, systems containing radioactive materials had been sealed, non-essential, non-radioactive systems had been removed, and regular radiation monitoring of the environment and of the radiological and physical state of the reactor was performed.

5.1. Clean up and restoration of the reactor building

As mentioned in section 1, from the late seventies until 1996, the reactor hall had been used for a number of chemical engineering experiments. Part of the equipment used in these experiments, primarily tanks, were left in the reactor hall and in the experimental basement. Most of this equipment had to be removed before the start of the project and the hall and the basement had to be cleaned up and repainted to ease decontamination, should these areas be contaminated during the project.

The reactor hall crane was checked and approved to lift up to 18 tons. The weight of the concrete cover at the reactor top is 15 tons. Even though the crane was checked initially, it still had to be repaired a number of times during the project.

The staircase to the top of the reactor had been removed during the chemical engineering experiments, but was re-installed for decommissioning. The outer railing around the top of the reactor was also re-installed. Further, the working space at the top of the reactor, which is quite limited, was extended by the erection of a scaffold with a working platform on top.

All drains from the floor of the reactor hall were routed to a tank in the basement. From this tank any spill could be transferred to the Waste Management Plant at Risoe.

The key system was changed, and new keys were issued only to a limited number of persons with permanent permission to enter the reactor building. Shoe barriers and monitors were arranged at the entrances to the DR- 2 area.

5.2. Safety documentation

When the DR-2 was closed down, it was categorised as a “reactor under decommissioning”, as mentioned above, and a revised “Safety Documentation for the DR-2” was prepared. At the same time the Danish regulatory authorities issued a revised “Conditions of Operation”, which forms part of the “Safety Documentation for the DR-2”. Since both of these documents were based on a sealed DR-2, and since the DR-2 project involves opening the reactor, new revisions to the documentation had to be prepared.

The safety documentation contains a description of the DR-2, the general safety rules applied, system descriptions, fire protection, accident analysis, radiation protection, organisation, administrative control and Conditions of Operation. It also contains six appendices covering radiation measurements around the DR-2, activity calculations, and material specifications.

The new safety document “Sikkerhedsdokumentation for DR-2” [3] is written in Danish and contain about 75 pages. It was prepared by a small drafting group and approved by the planning committee. The work on the safety document was started in the autumn of 1998 and finished during the spring of 1999. It was then submitted to the Safety Committee of the Risoe National Laboratory together with “Description of the DR-2 Project” [4]. After approval of this committee the two documents were sent to the Danish regulatory authorities together with a proposal for “Conditions of Operation”. The approval of the new “Safety Documentation for the DR-2” was received just before Christmas 1999 together with a revised version of the “Conditions of Operation”.

5.3. The DR-2 archives

In accordance with the “Conditions of Operation” for the DR-2 after its initial close down, two DR-2 archives with material of relevance for the decommissioning of the reactor were established, each contained in a locked steel cupboard. They were placed in two different buildings, initially in the Riso administration building and in the basement of the DR-2. They contained mainly folded paper drawings and system descriptions, arranged in file pockets. The technical archive of the DR-2 was also placed in the DR-2 basement in steel cupboards.

During the chemical engineering experiments there was a spill of chemical liquids in the reactor hall, which ran down into the basement and partly “showered” the archive cupboards, and damaged some drawings in the technical archive, but none of the DR-2 archive. Inspection of the technical archive also revealed that in three drawing cupboards where the

drawings were hanging down from cardboard strips and fixed to these by use of tape, the tape had not been strong enough. Thus some of the drawings had ended up at the bottom of the cupboards where they were soaked by the liquid spill. Fortunately, the damage was limited.

The requirement in the new “Conditions of Operation”, that the two DR-2 archives be identical, led to a thorough revision of both archives. At the same time, material relevant to DR-2, which had become available since the establishment of the archives, was included in them. In this way both archives were brought up-to-date. Also the drawing cupboards of the technical archive were examined and revised. Further the DR-2 archive and the technical archive in the DR-2 basement were moved to a safer place, but the two archives are still kept in different buildings.

It should be mentioned that while the two DR-2 archives contain much valuable information, no written record was found of where components were placed in the reactor when it was sealed. This led to some surprises when the reactor was reopened.

5.4. Other preparatory work

A concrete block facility was built in the reactor hall for the storage of drums containing radioactive waste produced by the project, and of radioactive components from the reactor. A movable hydraulic cutter was placed next to this facility such that radioactive components could be cut into pieces and dropped down into concrete lined waste drums.

A measuring facility, built of concrete blocks and lead bricks, was also constructed in the reactor hall. Here the γ -activity distribution along radioactive components could be measured when moving the component on a small carriage past a 5 cm wide lead collimator.

An attempt to get some idea of the magnitude of the activity of the components in the reactor tank was made. A shielding plug at the centre of the concrete lid on the top of the reactor was removed and a scintillation counter was placed on the top of the 5 cm lead layer below the plug. Assuming that all activity in the tank is situated at the centre of the core, the total activity was determined to be 60 GBq (about 1.5 Curie). The measurement also showed that the dominant radiation was from ^{60}Co , probably originating from activated stainless steel components, such as the regulating rod, the guide tubes and the control rods.

It is hardly surprising that ^{60}Co is the dominating radionuclide (cf. below). ^{60}Co may be produced by thermal neutron capture in cobalt impurities or by double neutron capture in ^{58}Fe . While the first process depends on the concentration in the reactor materials of the cobalt impurity, which may vary considerably, the latter depends only on the presence of iron which, in practice, will always be present in reactors, either as a construction material or as an impurity. For long irradiation periods, i.e. many years, it is a general experience that the double neutron capture in ^{58}Fe will be the dominating process. ^{60}Co may also be produced by (n,p) reactions in ^{60}Ni due to fast neutrons, but the neutron energy has to exceed 5 MeV, and the cross section is small.

The annual measurements on the surface of the DR-2 indicate that while ^{60}Co dominates the radiation field there seems also to be a contribution from radionuclides with longer half-lives, possibly ^{152}Eu .

A radiation survey at the outer surface of the primary circuit in the basement found radiation fields that are close to the background level.

The primary circuit was opened and a swipe test performed on the inside of one of the main coolant tubes. The activity found was very low. The most important radionuclide detected was ^{60}Co , but traces of ^{137}Cs , ^{152}Eu and ^{154}Eu were also observed. The surface contamination was of the order of 1 to 10 mBq/cm² for the radionuclides detected.

5.5. The reactor tank

It had been hoped that the reactor tank could be opened in January 2000, but due to the DR-3 leak, no manpower was available for the DR-2 project until May 2000. At this time the top concrete lid was lifted and placed on the floor of the reactor hall. Further, the two layers of 5 cm lead bricks were removed. Before the steel plates, which had carried the lead bricks, were removed, air samples were taken from the interior of the tank. These samples indicated no contamination of the air of either radionuclides or beryllium.

Next one of the steel plates was removed, and smear tests were taken from the wall of the reactor tank and from the surface of the beryllium reflector elements. No significant contamination was detected on any of the swipes. From the results of these tests it was concluded that the handling of components in the reactor tank would not give rise to contamination problems. Thus all of the steel plates were permanently removed. During this first phase the personnel working at the top of the reactor wore masks for protection from beryllium and radioactive dust. The use of the masks was discontinued, except when the beryllium reflector elements were taken out of the reactor.

A visual inspection of the interior of the reactor tank after removal of the steel plates revealed that there were fewer components in the tank than had been expected.

At this time the maximum radiation level at the reactor top was 450 $\mu\text{Sv/hr}$.

Before removal of components from the reactor tank started, the inner railing around the tank was re-installed. The removal of components from the tank involves the use of long “fishing tools”, applied by personnel standing over the open tank. Personnel were prevented from falling into the tank by the use of a safety harness fixed to a “gallows” mounted on the side of the reactor top. This gallows was installed as soon as the concrete lid was removed.

The opening of the reactor tank was followed by the removal, one by one, of all movable components in the reactor tank. Once out of the reactor, the radiation from and the γ -spectrum of the individual component were measured in a well-defined geometry. From these measurements the activity of identifiable radionuclides was determined. For long components the activity distribution along the components was also determined in the measuring facility of the reactor hall.

Initially it was the intention to temporarily store all the active components from the reactor tank in the storage facility in the reactor hall, and then to return them to the tank once a radiation survey had been made of the empty tank. However, permission was obtained to cut the active components into pieces, put them into waste drums and transfer the drums to a storage facility at the Risoe Waste Management Plant.

The first components to be removed were the five shim-safety rods. They were hanging down from the bottom of the guide tubes, which again were situated in the storage rack at the tank wall. The safety rods were lifted out of the guide tubes by the use of one of the electromagnet rods, which during reactor operation were used to carry the shim-safety rods.

Once out of the guide tube the lifting was taken over by a special lifting fork. After the activity measurements, the top part of the shim-safety rods, i.e. the shock absorber and the magnet armature, was cut off and deposited together with the absorber part of the rods in a waste drum. There was no cutting of the absorber part of the rod in order to avoid possible release of activity from the boron carbide powder in the rods.

The removal of the rods gave rise to a few problems, none of which were serious. One of the guide tubes was bent at the top so that the shim-safety rod (SR(A)) could not be lifted out of the tube by use of the magnet rod. Therefore the two components were taken out together and separated by cutting. In another case a shim-safety rod (SR(B)), with its guide tube on top, had got stuck in a shim-safety rod fitness gauge. The gauge with the shim-safety rod inserted was placed in the storage rack. It turned out that, with the use of a limited amount of force, the guide tube and shim-safety rod were separated from the gauge.

Next the remaining five guide tubes, three grid-plate plugs and the test equipment, all located in the storage rack, were taken out, cut and deposited in drums. Figure 2 presents the collimated radiation level measurements made in 1 meter distance from the six guide tubes along their length. Note that since the amount of stainless steel in the guide tubes is not constant along the tubes due to the bottom flange and the holes in the tubes, the radiation level is not proportional to the integrated neutron flux. The grid plate plugs, which were made of aluminium and used to plug unused positions in the grid plate, were provided with threaded holes in which a total of 24 stainless steel screws were placed. These screws had been used to attach the guide tubes to the special fuel elements.

The maximum radiation level at the reactor top had been reduced to 300 $\mu\text{Sv/hr}$ after the removal of these components.

The removal of the guide tubes was followed by the removal of the regulating rod. Its absorber part was by far the most active component in the tank. In fact it was so active that it was not possible to make a direct radiation measurement in the reactor hall, since the γ detector used was overloaded, even at the largest possible distance. After its absorber part had been cut into two parts and dumped into a concrete-lined waste drum its activity was measured.

After the removal of the regulation rod the radiation level at the reactor top had been reduced to 150 $\mu\text{Sv/hr}$.

Due to the high radiation level around the drum with the regulation rod, it was decided to cut a number of very low activity components in the tank into pieces and to deposit them on top of the regulation rod in the drum to reduce the radiation level. The components deposited in this way were the five magnet rods (except the top end, which contained rubber cables), the carrier rod of the regulation rod, and six rods used to prevent upward movements of the special fuel elements and guide tubes. The aluminium V-tube facility on the grid plate was removed, measured, cut and deposited in a waste drum.

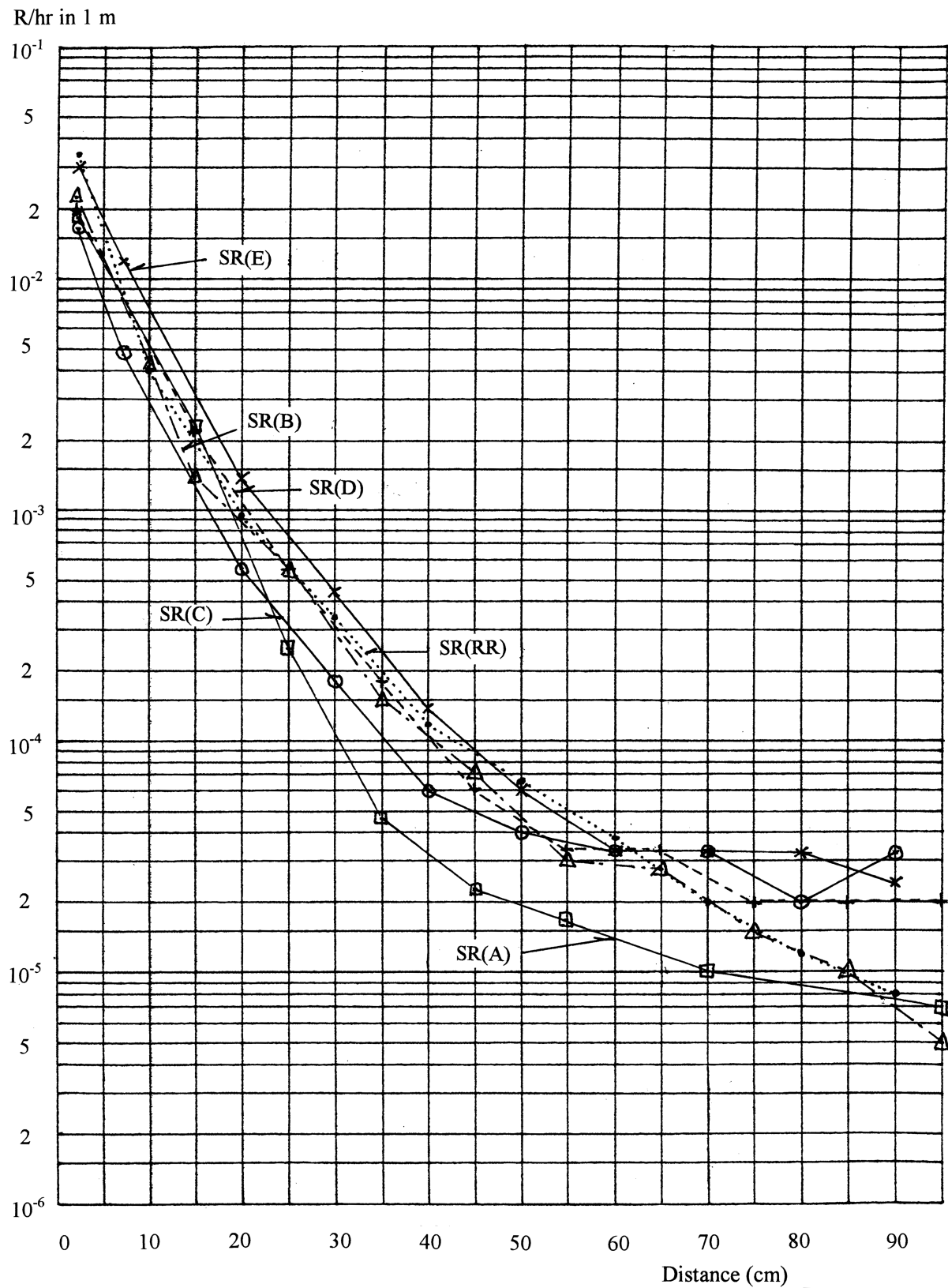


Fig. 2. Collimated radiation level in 1 meter distance along guide tubes versus distance above bottom end of tube. SR (X) is the guide tube of the X's shim-safety rod, SR(RR) is the guide tube of the regulation rod.

So were an aluminium “cooling chimney” from a vertical hole in the thermal column, which initially was intended for a pneumatic tube, and two aluminium “water holes”, used to modify the flux distribution of the core.

Only the removal of the last water hole from the grid plate caused some difficulties, since it could not be removed by hand. A special tool had to be used which applied a force of 60 kg.

Components from the reactor tank, which had low or no activity were put in plastic bags and stored in the reactor basement. For unknown reasons, when DR-2 was initially shut down, a 20 cm diameter hole had been cut in one of the steel plates at the reactor top using a cutting blowpipe, and the cut-out steel disk had dropped down on the grid plate. This non-radioactive plate was removed by use of a strong magnet.

The last components to be removed from the tank were the 12 beryllium reflector elements. They consisted of a aluminium lower part, which was inserted into the grid plate, the beryllium reflector part, and a top part. According to available drawings the top part was made of aluminium, but it turned out that it was actually made of stainless steel. It was noticed that the surface of the beryllium elements was shiny, and there was, as expected, no indications of corrosion. The reflector elements were lifted up, put into plastic bags and stored in two stainless steel containers, six in each. Measurements of the activity of the reflector elements were only performed on two elements.

Even though “fishing tools” were available for the removal of the majority of the components in the tank, in most cases simple hooks were used.

After the removal of the reflector elements the radiation level at the top of the reactor was 75 $\mu\text{Sv/hr}$.

As the waste drums were filled up, they were transported to the Waste Management Plant. The same was true for the two stainless steel containers with the beryllium reflector elements.

At this point all movable components in the reactor tank had been removed. TL dosimeters were placed at various locations in the tank to determine the radiation fields. These measurements are now being analysed.

A few samples of activated materials were taken and will be handed over to the Waste Management Plant.

5.6. Activity of components removed from the reactor tank

As mentioned in section 5.5 the activity of almost all components taken out of the reactor tank was measured. The results obtained are shown in Tables I and II.

From Tables I and II it is seen that ^{60}Co is the dominating radionuclide, and that the major part of the total γ -activity is contained in the regulating rod. Further the stainless steel guide tube also contributes significantly to the total ^{60}Co -activity. These results were expected. What is more surprising is that the beryllium reflector elements contribute significantly to the ^{60}Co -activity and that about half of the ^{60}Co -activity of these elements seems to come from the beryllium part. This must be due to impurities in the beryllium.

It is also of interest to note that the ^{137}Cs -activity has been found primarily on the beryllium reflector elements, even though a small amount of activity has also been detected on the V-tube facility. This seems to indicate that beryllium metal surfaces are more likely to adsorb cesium than aluminium and stainless steel.

The total activity of the removed tank components, about 36 GBq or 1 Curie, agrees quite well with the early estimate of 1.5 Curie activity in the tank. The remaining activity in the tank, presumably of the order of 10 to 20 GBq or 0.25 to 0.5 Curie may well be present in the lead nose of the thermal column and possibly also in the lead shield around the instrument thimbles. There are a few remaining stainless steel parts in the tank, but they are unlikely to contain all of the remaining activity.

5.7. The hold-up tank room

The next room to be opened was the hold-tank room in the basement of the reactor hall. The entrance to this room had been closed with concrete blocks, which were now removed.

Stored inside the room, in addition to the hold-up tank, were five stainless steel boxes with radioactive components from the hot-cell facility and other components, primarily from the DR-2.

The five stainless boxes were taken out and sent to the Waste Management Plant storage facility. With two exceptions all other components in the room were taken out, and divided into the following four categories according to their measured activity:

- A. Non-radioactive parts that could be treated as ordinary waste
- B. Parts that perhaps were slightly radioactive and that should be measured in more detail in another building.
- C. Slightly radioactive components that were marked, put into plastic bags and stored in the basement.
- D. Radioactive parts that were stored in the storage facility in the reactor hall for later cutting and storage in drums.

The two exceptions were a heavy box containing electromagnet(s), which were presumably used in a reactor beam experiment, and a heavy piece of equipment, possibly a shielded chopper. Parts of these items were radioactive and, due to their weight and size, difficult to move. The origin of these pieces is being investigated in the hope that drawings of them can be found to make dismantling easier.

5.8. The igloo in front of the thermal column

One of the concrete blocks in front of the igloo of the thermal column was removed early in the project in order to get some idea of what had been stored inside the room. While less than expected had been stored in the reactor tank, considerably more than expected had been left in the igloo. A number of the items have been identified, e.g. V-tubes, an underwater camera, beam plugs, ion chambers, a car for removal of the plugs in the instrument thimbles, graphite stringers etc.

The removal and measurements of the components stored in the igloo has been started, but it is far from finished. The components taken out will be measured by use of a hand monitor and divided into the four categories listed in section 5.5.

Table I. ^{60}Co activity in tank components

	Absorber part (MBq ^{60}Co)	Guide tube (MBq ^{60}Co)	Total (MBq ^{60}Co)
Regulation rod SR(RR)	23.000	871	24.000
Shim-safety rod SR(E)	114	741	855
Shim-safety rod SR(D)	75	558	633
Shim-safety rod SR(C)	122	500	622
Shim-safety rod SR(B)	106	614	720
Shim-safety rod SR(A)			918
Grid plate plug (6 screws)			36
Grid plate plug (9 screws)			51
Grid plate plug (9 screws)			43
V-tube facility			23
Water hole 1			2
Water hole 2			44
Beryllium element No. 60 (SS top part of No. 60: 235 MBq ^{60}Co , four SS screw: 56 MBq ^{60}Co)			809
Beryllium element No. 58			770
<hr/>			
Total (including all 12 Be-elements)			38.000

Table II. ^{137}Cs activity in tank components

	(MBq ^{137}Cs)
Beryllium element No. 60	49
Beryllium element No. 58	43
V-tube facility	2
<hr/>	
Total (including all 12 Be-elements)	550

6. Pending issues

Even though the reactor tank has been emptied for all removable components, and the radiation level in the tank has been decreased considerably, there is still a significant level whose origin has to be determined.

The origin of the two remaining heavy components in the hold-up tank room has to be determined as do procedures to take them apart.

The removal of components from the igloo of the thermal column has to be completed and decisions on the fate of the components have to be made.

7. Future work

Once the igloo has been emptied graphite stringers of the thermal column will be pulled out of the thermal column and their activity measured as a function of the distance to the core. It is hoped that graphite stringers can be removed all the way to the outer side of the lead thermal shield so that a sample of the lead can be obtained.

It is the intention to open some of the beam tubes and extract their beam plugs. The activity along the plugs will be measured and so will the activity of the concrete around the plugs. Some of the S-tubes (irradiation tubes) will also be opened and examined.

Measurements of the radiation level in the concrete will be made in vertical channels in the concrete shield by use of TL dosimeters.

Some holes will be core drilled through the concrete at the level of core. The activity of these cores as a function of the distance to the core will be measured to determine the extent of the activation of the concrete shield.

Further measurements will be made of the contamination in the primary circuit by opening it in a number of places.

It is hoped that these activities can be finished before the end of the year, depending on the availability of the necessary personnel to carry out the work.

Initially it was part of the DR-2 Study Project to consider how the reactor could be dismantled. This task is still relevant, but due to the recent decision to decommission all nuclear facilities at Risoe National Laboratory the planning of the dismantling of the DR-2 will have to be part of the overall planning of the Risoe decommissioning programme.

8. Lessons learned

So far the project has progressed without major difficulties, and few unpleasant surprises have been experienced. The main problem has been that the needed manpower has not always been available due to the understandable fact that work on DR-3 has had first priority. If anything there has so far been a tendency to overestimate rather than underestimate the difficulties, e.g. the handling of beryllium and potential contamination.

Nevertheless a number of lessons have been learned during the project. These lessons are neither very original nor unknown, but they are nevertheless important.

It is important to keep good records of all activities undertaken including information on what has been done, which results have been obtained and where have the various components gone.

It is important to appoint one person to be responsible for the archives of the reactor. This person has to ensure that the archives are kept up to date and that new, relevant

information is included. This is of particular importance if the archives have to be kept for a long period of time.

Small samples of typical reactor materials should be taken and stored so that later on when the ^{60}Co activity has decayed long-lived radionuclides can be identified.

It is important at an early stage to decide on the type of container to be used for the final disposal of the radioactive waste. This container should be used once removal of radioactive components is initiated so that later reloading from one container type to another will not be needed.

It is important to have officially accepted release criteria so that the amount of radioactive waste for final disposal can be kept as small as possible.

Before the actual dismantling of nuclear facilities is undertaken, it is important to have a repository for radioactive waste available. The alternative is to build an interim storage facility, but this solution will increase the cost of the decommissioning and the personnel dose.

It is important to carry out the decommissioning safely and economically. It is easy to use a lot of extra funds without improving the safety significantly. The best can easily become the enemy of the good.

ACKNOWLEDGEMENT

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Approach to decommissioning planning of the Egyptian research reactor ET-RR-1

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Abstract. Based on the broad inspection programme carried out for ET-RR-1, it has been clearly demonstrated that the whole reactor system, subsystem and components need urgent renewing, replacement, maintenance and testing; (in order to increase the plant lifetime for another 5–10 years). Intensive studies and evaluation of different strategies for future planning to update the current reactor systems, upgrading reactor power or decommissioning are under investigations with planning for the future decommissioning as part of the programme.

1. Introduction

The first Egyptian research reactor ET-RR-1 is a 2 MW thermal, tank type reactor. The reactor is of the WWR type. It was designed for isotope production as well as in and out of core experiments. It went critical for the first time in the fall of 1961 [1].

As aging has not only affected system availability, but also reactor safety, it was felt necessary to carry out a broad-based inspection of the whole facility, since its age is approaching 40 years. To perform such an investigation, the fuel, being the main radiation source, should be removed from the system (reactor and storage pool).

Considering the amount of the spent fuel assemblies accumulated in the storage pool and those that will be produced if the operating schedule is maintained, it was obvious that the present storage capacity should be extended. This necessitated the design and installation of a New Storage Facility (NSF), [2]. The site was selected taking into consideration the relevant aspects (infrastructure facilities, minimum transportation distance, safeguards issues, etc.), Figure 1. This need emerged at that stage to solve the problem of spent fuel storage in a safe manner with sufficient capacity to allow for future decommissioning of the ET-RR-1.

The NSF was designed and constructed for wet storage of vertically arranged fuel assemblies in storage rack units arranged inside the storage tank. By including provisions for encapsulation and fuel transportation in the facility, NSF ensures that the facility is prepared for further interim or long term storage before the final disposal of spent fuel assemblies Figure 2.

2. Project objectives

One of the main objectives of this project is the inclusion of decommissioning requirements in the design of the new spent fuel storage facility (Section 3), as well as planning for decommissioning of the entire reactor facility (Section 4). Relevant, future milestones in the reactor's life cycle include:

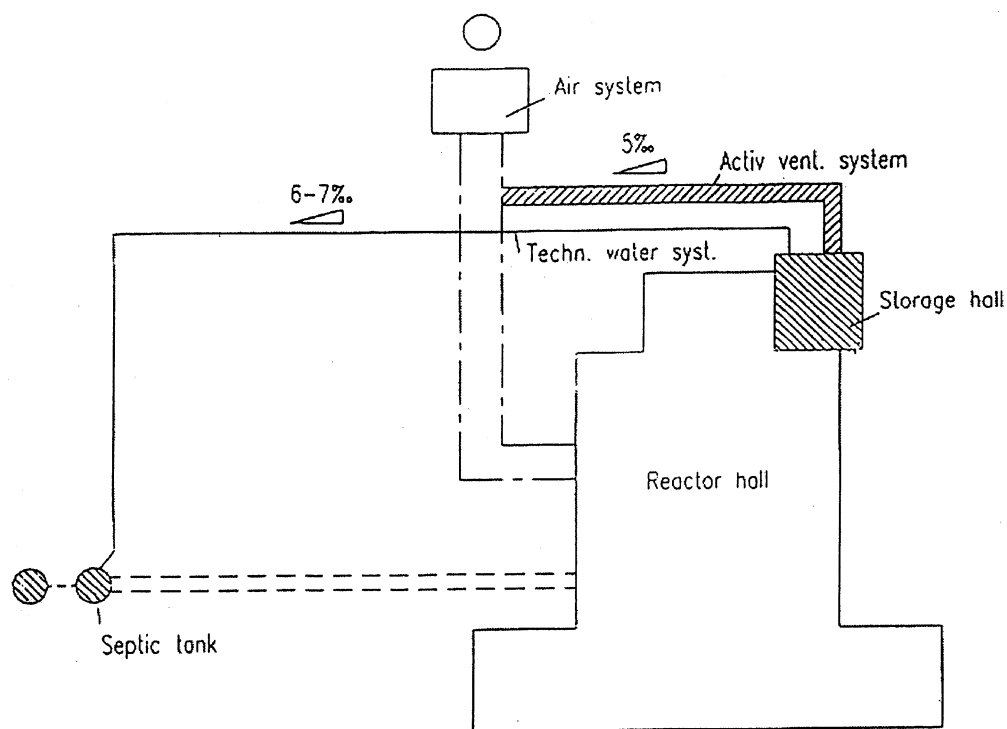


FIG. 1. General plan.

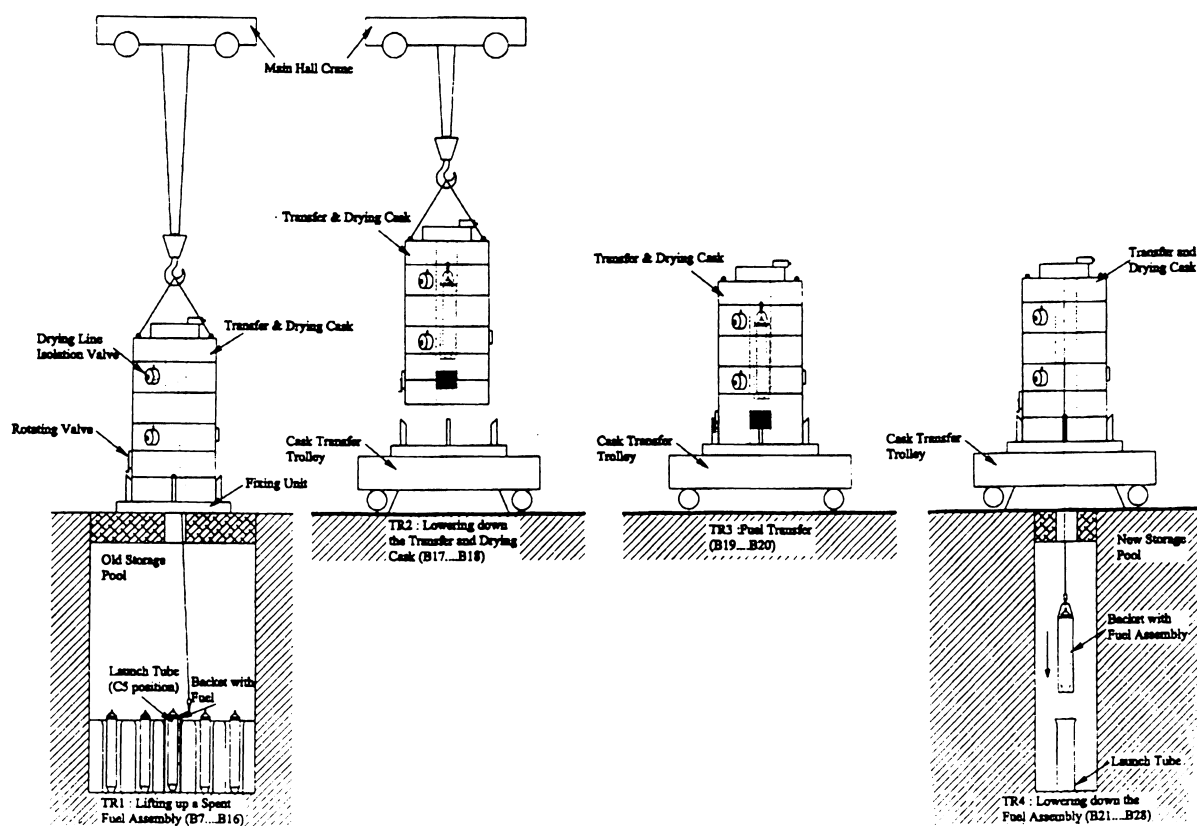


FIG. 2. Flowchart of the spent fuel transfer procedure.

1. The upgrading of reactor systems for safe operation until all fresh fuel stocks have been burned up (exhausted) taking into consideration the decommissioning plan.
2. The upgrading of the reactor power with all necessary modification and changes in the core, new fuel type, reflector, mechanical, cooling and safety systems, etc., to extend the operational lifetime at the higher power with future decommissioning considered in the redesign.
3. Shutdown the reactor and implement the decommissioning plan.

Decisions depend on the interaction of complex factors such as the systems technical condition, utilization plan, economical considerations and management and organization system in addition to technical staff qualification, training and responsibilities.

3. Design features to facilitate decommissioning

The need to consider ultimate decommissioning of the plant facility has led to the incorporation of the following design principles:

3.1. Fuel handling equipment and transportation facilities

A clear route must be provided through the necessary containment, shielding and handling facilities for the removal of the fuel from the storage facility. This objective is achieved in the design of the fuel handling operations, which allow for the recovery and transfer of the fuel, via a Transfer Cask. The level of safety provided by those components is more than adequate for the fuel removal operations.

3.2. Fuel recovery

The Fuel Storage Pool (or in case of encapsulation, the Storage Tubes) provides the main containment boundary for the fuel assemblies during the storage period. The fuel is to be stored in an environment that will ensure that the fuel condition remains adequate for unloading, handling, transport and subsequent storage. This objective is achieved by control and monitoring of the parameters of the fuel storage environment. This includes monitoring of the water, or (in case of encapsulation) control of the inert gas that surrounds the fuel during storage in the storage tube.

As a general rule, during the design stage of the NSF, decommissioning considerations are taken into account such that, when needed, decontamination and dismantling of structures and equipment, together with transportation of spent fuel and wastes should be facilitated. The quantities of waste can be minimized, and occupational exposures kept as low as reasonably achievable (ALARA). These design provisions are consistent with the safe and efficient operation of the facility.

Design features to assist decommissioning include the following:

- Provisions specifically intended to facilitate the removal of equipment and systems during the decommissioning stage; and
- Planning and arranging the NSF and related operations such that the contamination of those areas and equipment which are not likely to be readily decontaminated is minimized as far as practicable. Examples of these provisions referred to above include minimizing penetrations through pool walls for piping systems, etc.

3.3. Contamination control

Any areas of contamination within the plant should be restricted and controlled in order to prevent spread of radioactivity and to minimize the amount of waste arising. Design provision provides clear and minimized routes for the removal of contaminated items. These objectives are achieved by a design philosophy that limits the potential for contamination. One of the design criteria incorporated into the NSF is the containment and control of contamination. The containment system consists of engineered barriers, which are further protected by a HEPA filtered ventilation system.

3.4. Results of reactor facility inspection

Widespread inspection plans have been carried out for all reactor systems. This was accompanied by the renewal, replacement, maintenance and testing of subsystems and components in order to increase reactor safety and reliability. The conclusions led to intensive studies of different strategies as follows:

- (1) Update reactor systems for safe operation until all fresh fuel stocks being burned up (exhausted) taking into consideration the decommissioning plan.
- (2) Upgrade reactor power with all necessary modifications and changes in the core, new fuel type, reflector, mechanical, cooling and safety systems, etc., to extend the operational lifetime at higher power taking decommissioning into account during the redesign
- (3) Shutdown the reactor and prepare the decommissioning plan.

The decision depends on the interaction of complex factors such as the technical condition of the various systems, utilization plan, economical considerations and management and organization system in addition to technical staff qualification, training and responsibilities. Defining the objective of the plan and ensuring flexibility and dynamic action towards fulfilling its success will enable the objective to be met [3].

3.5. Decommissioning plan

Decommissioning requirements should be kept in mind during the whole operational life of the plant. This means that up to date documentation should be kept and modifications should be recorded to facilitate the planning of decommissioning. The decommissioning of the reactor facility has been divided into four main practical stages. These are:

- (1) Planning and licensing
- (2) Defuelling
- (3) Decontamination
- (4) Demolition

The four stages of decommissioning are described below.

A report list of research reactor decommissioning projects can be found in Appendix I of Reference [4]. The document also contains data about the projects, some of which might be applicable in the design of the decommissioning of the Spent Fuel Storage Facility.

3.6. Planning and licensing

The detailed decommissioning plan will be submitted to the Egyptian regulatory authority. The operating organization prepares the decommissioning plan to ensure safety during decommissioning and thereafter. It should be submitted for review and approval by the regulatory body before decommissioning activities are begun, [5]. This plan should be updated during facility operation if any operational states or problems are identified that

impact on it. It should include an evaluation of one or more appropriate decommissioning alternative plans for the facility. The operating organization may contract decommissioning tasks to other organizations, but not its responsibilities. The regulatory body should be notified of and agree to the delegation of these tasks. The operating organization has to ensure that any contractor performing these tasks complies with the requirements of the approved decommissioning plan.

The responsibility of the operating organization shall be terminated only with the approval of the regulatory body. Normally this responsibility lasts until the regulatory body gives approval to release the site for unrestricted use or to transfer the responsibility to another organization. It is possible that, another organization may take over from the former operator, for the specific task of planning and implementing the decommissioning. In this case, the decommissioning organization assumes the operator's responsibilities.

If the Spent Fuel Storage Facility remains operational after decommissioning of the reactor facility, it should be assured that the common services are maintained. Spent fuel storage facilities should be considered to be operational until all the spent fuel has been removed. After the stored spent fuel has been removed, the facility can be further decommissioned by removal of residual radioactive contamination and facility dismantling, as provided for in the approved final decommissioning plan.

There are various options as alternative approaches to decommissioning such as:

- Protective storage in an intact condition after all fuel is removed from the reactor and stored in the NSF,
- Entombment of radioactive structures and large components after removal of readily removable components and wastes,
- Removal of all radioactive materials and thorough decontamination.

3.7. Defuelling

The majority of the radioactive inventory in the Spent Fuel Storage Facility is bound up in the stored fuel. When the arrangements have been made to transport the fuel assemblies away from the storage facility, each individual fuel assembly is removed from the storage (using the existing fuel handling equipment).

Any appropriately licensed cask can be utilized for fuel assembly off-site transfer to the final repository.

3.8. Decontamination

After fuel handling and transportation of other radioactive components, the structure and equipment should be surveyed, clearly predicting contaminated areas. Decontamination technical procedures with proper equipment should be prepared, controlled and supervised by specialists during the decommissioning process of the facility and that the contaminated areas should be restricted and minimized. The fuel handling tools are readily accessible for decontamination control, both during service and decommissioning phases.

3.9. Demolition

The potential to employ conventional demolition techniques must be maximized for both the non-contaminated and contaminated mechanical/electrical plant, as well as the civil structure.

This objective is achieved by the conventional nature of the plant, the building design, and its construction. There are no special or hazardous materials used which required the use of expensive or exotic dismantling techniques.

4. Radiation protection and safety control

4.1. Radionuclide inventory

Following approval of the decommissioning plan the radioactive characterization of the storage and the licensed site can take place. The site characterization survey should assess the radioactive inventory of the storage, as a pre-requisite to defuelling. After the defuelling and decontamination stages a final radioactive release survey should take place. The results of the final release survey should be submitted to the regulatory authority as part of the application for release of the nuclear site license. The demolition of the remaining uncontaminated structures can then be completed.

A radiological program is necessary during decommissioning. For each task, radiological protection measures should be planned and implemented. Normally the first step is to estimate the radionuclide inventory then to plan the decommissioning activities to ensure that exposures comply with the ALARA principles.

The next step is to execute the program, with close monitoring of the radiation fields and exposures. The final step is to make a survey on the completion of decommissioning in order to demonstrate that the required radiological condition of the spent fuel storage pool has been achieved.

A major factor to be taken into account when deciding the method and extent of decommissioning is the estimated quantity of radionuclides present and the nature of their principal physical and chemical forms. The half-lives of the radionuclides that are present in significant amounts are important in determining the length of time for which various decommissioning activities might be deferred.

Apart from the spent fuel, the radionuclides inventory can be divided into two categories:

- (1) Activation of the structural materials, and
- (2) Surface contamination.

The contamination may consist of activated corrosion products, and / or fission products. Normally the radionuclide of most concern, which may cause high radiation fields for several years after shutdown, is ^{60}Co , primarily as a result of the activation of stainless steel parts (bolts, nuts, etc.). Knowing the composition of the elements of the storage pool, the inventory can be estimated reasonably by calculation. For confirmation, statistically significant samples and measurements should be taken. Surface contamination may be evaluated by direct measurements and using swabbing techniques.

4.2. Hazard and risk assessments

The decommissioning plan should take into consideration the possible hazards and associated risks involved, and include all steps that lead to eventual complete decommissioning to the point that safety can be ensured with minimum surveillance. These stages may include storage and surveillance, restricted site use and unrestricted site use.

The preparation of a hazard assessment should take into account the radiological dose estimates and the impact of conventional hazards.

Conventional hazards must also be assessed to estimate the impact on safety. These include the release of toxic or corrosive materials, industrial hazards, man-made events and intrusion. Some events can result in an increased radiological hazard. The most likely hazards and initiating events include airborne activity as a result of the failure of engineering features (e.g. loss of ventilation / filtration).

4.3. Safety management and monitoring

Application of a well defined decommissioning strategy and plan, with careful attention given to the ALARA principles, will result in the reduction of radiation sources, dose rates and worker time in radioactive zones. For instance, the sequence in which decommissioning activities are conducted may have a significant impact on the doses received by workers during the decommissioning and waste handling operations.

Minimization of the radiation exposure to personnel requires a system of radiological control procedures. The system includes the establishment of controlled access working zones; issuance of radiation work permits; use of protective clothing and respiratory protection; control of storage and prompt removal of all dismantled components and parts; and surveillance and monitoring of health physics. Remote tools and shielding techniques may also be used to reduce radiation exposure to personnel.

Monitoring includes continuous measurement of the radiation levels in the work areas; recording and assessment of exposure to personnel during and after the work; surveys of surface contamination and monitoring of the air contamination levels. Additional monitoring may be required during certain operations (i.e. during the removal of concrete). Techniques to monitor personnel may include hand and whole body monitors; alarming dosimeters; whole body counting; bioassay and film dosimetry.

Procedures for exposure control to ensure that the individual doses resulting from decommissioning activities remain within the authorized limits should be implemented. All the workers participating in decommissioning activities (including managers, supervisors, Quality Assurance personnel, etc.) should be trained in radiation protection before beginning the operations. Personnel monitoring should control the individual dose.

Discharge of radionuclides via airborne and liquid pathways should be controlled, monitored and recorded to demonstrate that they are within the authorized limits. The off-site monitoring programs that existed during operation could be valid, if necessary with minor changes, to the conditions existing during decommissioning.

Training is a very important tool in achieving lower radiation exposures. Training of workers should cover radiological and conventional hazard and should be completed well before work commences. Supervisors and other key workers should be experienced in radiation protection and be familiar with the spent fuel storage facility.

During the planning and execution of the decommissioning work experts should be available for consultation and to assist in training.

4.4. Personnel dose uptake

The potential for access during dismantling operation must be maximized, with the recognition of the need to limit the exposure to personnel to the lowest reasonable level (ALARA). This objective is achieved by the virtue that, once the fuel has been removed from the storage facility, there are no significant radiation hazards left in the facility. By the careful use of contamination control techniques, the dose to operators during the decommissioning and dismantling stages will be very low. Occupational doses can be estimated from radionuclide inventory, contamination level and radiation field data. Estimates should be prepared for each of the work packages, taking into account the distance from the radiation source and the time required completing the activity.

Non-occupational doses must also be assessed and should be based on source terms and exposure pathways. Experience with the decommissioning projects so far indicates that doses to the general public are very small.

In order to decommission the contaminated items, a temporary containment area should be established. This area could be formed by local tenting techniques: the contained volume would be exhausted by mobile ventilation/HEPA filtration equipment. The necessary conventional tools, decontamination equipment and health physics monitoring instrumentation can be located within the containment area.

The use of standard decontamination procedures (such as vacuum cleaning, washing or swabbing techniques) can be used to achieve acceptable contamination levels. The final means of packaging and disposal would be influenced by what could be economically achieved by decontamination, taking account of operator dose uptake.

A health physics program, in conjunction with contamination control techniques, ensures that the dose to operators during decommissioning activities will be minimized. Following health physics clearance, all waste materials will be segregated and routed out of the spent fuel storage facility. The common route for all waste ensures that appropriate controls and associated quality assurance procedures can be correctly applied.

4.5. Estimated quantity of active waste produced during decommissioning activities

The major source of active waste arising during decommissioning activities will be due to the dispatched fuel assemblies. This activity will take place both within the storage pool and in the fuel storage during the storage period and during eventual fuel removal from the storage facility.

Other potential sources of contamination include liquid waste arising from the pool cleaning system, equipment (cask) decontamination prior to fuel transfer, dismantling of the active ventilation system and services ducts and piping; following the defuelling process. The quantity of active waste associated with these operations is expected to be extremely low, in comparison to the potential fuel activity levels.

The dominant isotope within the fuel after, for example, 20 years storage will be ^{90}Sr and ^{137}Cs . The total activity of these isotopes is estimated to be approximately 3.6×10^{12} Bq per assembly.

The decommissioning of a Spent Fuel Storage Facility (approximately 175 stored fuel assemblies) results in the removal of a total activity as a result of decommissioning activities of about 6.3×10^{14} Bq.

The drying system internals will be checked for any remaining activity following the initial decontamination. The equipment will then be disposed of by methods appropriate to the activity level detected.

One option to consider is dry storage for the longer term, because of its inherent passive nature and low operating costs. Dry spent fuel may also be more amenable to conditioning for disposal. This may be achieved at a later stage, by applying the dry storage technology that can be used for the second reactor on the Inshas site. In any case, before dry storage of the fuel, generally about 10–15 years of prior cooling due to temperature / strain limits are necessary.

4.6. Estimated operator dose uptake during decontamination activities

Decommissioning of the Facility will result in personnel dose uptake from two operations, namely fuel assembly unloading and plant decontamination.

The fuel unloading operation includes all the operations associated with normal fuel loading, but with the omission of fuel drying. It can be assumed that the fuel assembly unloading operations will take place after 20 years of fuel storage.

During facility decontamination activities the major source of dose uptake to the operators will be due to active fuel crud, which has become distributed throughout the confinement system of the Spent Fuel Storage Facility during the handling of fuel assemblies. Other potential sources of contamination will include water from the overflow of the pool, decontamination, dismantling of the active ventilation system and dismantling of the service piping.

Most decontamination activities will take place after defuelling. The policy of limiting personnel exposure to levels, as stated by ALARA, is achieved through recognition that the most significant dose hazard to operators is the fuel itself. By use of careful contamination control techniques, such as the provision of temporary shielding where necessary and limitations to operator proximity to contaminated areas, the dose uptake to personnel due to decontamination activities will be kept low, in comparison with the dose uptake during defuelling.

4.7. Records of information important to decommissioning

Records of operating information important to the safe and effective decommissioning of the facility should be maintained. These records should include the following:

- (a) Records of contamination occurrences that might affect decommissioning, including the nature and extent of the contamination in and around the facility, when: residual contamination remains after any clean up procedures; or there is a likelihood that contaminants may have spread to inaccessible areas.
- (b) As-built drawings of:
 - (i) Equipment and structures in restricted areas where radioactive materials are stored.
 - (ii) Locations of inaccessible contamination, such as buried pipes, which may be subject to contamination.

4.8. International work in the subject area

Although over a hundred nuclear installations have been, or are being decommissioned so far, generally there are no accepted and recommended procedures to deal in a fully optimised manner with nuclear reactors or spent fuel storage facilities at the end of their service life. Technical experts agree that sufficient experience has been gained to demonstrate that such dismantling can be carried out, without unacceptable impact on man or environment. Even though progress has been made in the development of the technology and methodology of decommissioning, further work is required to improve equipment and techniques, reduce costs and exposures, or in one word to achieve an industrial scale for the whole process.

Information from decommissioning projects on power reactors and other nuclear facilities can also be adapted. However, it is recommended that a thorough review of the available technologies and processes be made before the commencement of the preparations for decommissioning.

Various IAEA and other international publications deal with the subject of classification of the decommissioning stages. For the sake of common understanding it is agreed to speak about three stages defined in [6].

These definitions should be interpreted as being specific to individual research reactors, but may be applicable to a wet storage facility as well.

- (a) Stage A (storage with surveillance): the first contamination barrier (the pool itself) is maintained as it was during operation, but the mechanical openings are sealed permanently. Surveillance, monitoring and inspections are carried out to ensure that the facility remains in good condition.
- (b) Stage B (restricted site release): the first contamination barrier is reduced to a minimum size by removing the easily dismantled parts. Sealing of the barrier is enforced by physical means and the biological shield is extended, if necessary, to surround the barrier completely. After decontamination, the building of the fuel storage may be modified or removed if it is no longer required for radiological safety. Access to the building may be permitted.
- (c) Stage C (unrestricted site use): all materials, equipment and parts of the Storage Facility still containing significant levels of radioactivity are removed. The site used by the Storage Facility is released for unrestricted use. No further inspection or monitoring is required.

It is necessary to mention that the above listing is a tentative list, the decommissioning does not necessarily require the adoption of all stages in a stepwise or continuous process. It may be possible to permit unrestricted use of the site while other parts of the complex area are still subjected to restriction. Equally, there are possible variations within any individual stage. As a general rule, it may be stated that if a facility is used or reused for the fuel assemblies: only trace activity is expected in the storage positions after fuel removal. Any loose contamination after unloading can be removed by underwater vacuum cleaning techniques. The pool walls and the storage tubes are designed to be corrosion resistant to the external environment and to maintain their containment boundary for the life of the storage.

The fuel handling tools are readily accessible for decontamination control, both during service and decommissioning phases.

5. Conclusion

Inspection of ET-RR-1 clearly demonstrated that:

- the reactor tank, core components and other mechanical equipment are in good shape and can be utilized for at least 10 years;
- the available fresh fuel (EK-10) is enough for more than 10 years of operation, following the operation scheduled in last decade; and
- reactor instrumentation for safety, control and process system should be updated and renewed for reliable and safe operation.

Lessons learned from these investigations are the following:

It is possible to extend the lifetime of the reactor and even increase the power level. However, while doing this one must take into account the requirements of future decommissioning of the plant. The changes must consider decommissioning planning and any necessary equipment needed to achieve future decommissioning.

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Generation of database for future decommissioning of CIRUS

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Abstract. Safe decommissioning of a research reactor in a planned manner is inevitable at the end of its useful life even after refurbishment and life extension. This involves advance planning, adopting state of the art technology, development of required new technology, a well thought out plan for nuclear waste management and necessary research and development in the areas of decontamination to recycle and reuse most of the metallic materials. The 40 MW thermal research reactor CIRUS at Bhabha Atomic Research Centre, Mumbai, India is being refurbished after 37 years of operation. Several part-decommissioning activities were carried out during the refurbishment. This was also the right time and state of the reactor to generate the necessary data and document the experience gained and lessons learned to aid in the planning for future decommissioning of CIRUS. This report presents the details of radiological mapping and characterization studies carried out, experience gained in cleaning/decontamination, dismantlement works carried out for repairs/replacement of structures, systems and components and development of new devices/techniques. It is expected that this work would considerably aid in working out an appropriate strategy of decommissioning of CIRUS when needed in the future.

1. Introduction

CIRUS, a 40MW(th) tank type, natural uranium fueled, heavy water moderated and light water cooled research reactor with a maximum thermal neutron flux of 6.5×10^{13} neutron/cm²/sec, is located at the Bhabha Atomic Research Centre, Mumbai, India. The reactor was commissioned in 1960 and has operated successfully for over 37 years. The reactor has been extensively used for Isotope production, neutron beam experiments, neutron activation analysis, research, material testing, fuel development, training etc. CIRUS has shown signs of ageing since 1990 as was evident from the reduced availability of the reactor. Systematic ageing studies revealed the need to refurbish the reactor for life extension and accordingly the reactor was shut down during Oct. 1997 for refurbishment activities, after receiving the necessary approvals from safety authorities.

1.1. Objective

The objective of the project was to effectively utilize the experience and expertise gained during the planned refurbishment of CIRUS to generate an extensive data base which would be useful for detailed planning and optimization of the decommissioning process to be executed in a safe and economical manner as and when the reactor is finally shut down in the future. The project objective was also in line with the IAEA CRP titled “Decommissioning Techniques For Research Reactors”. Accordingly, the following areas were included in the scope of our project namely

- Dismantling experiences and expertise gained (Primary coolant piping and heat exchangers)
- Radiological characterization planning and evaluation (Reactor structure and primary coolant system components)
- Decontamination planning (sumps, concrete structures, primary coolant system valves and heat exchangers)

- Development of remotely operated equipment (Reactor hall crane, spent fuel bay cleaning and reflector graphite sampling)
- Waste management (Large quantities of primary coolant piping, heat exchangers and contaminated soil)

2. The reactor

The reactor core is housed in a vertical cylindrical aluminium vessel with aluminium lattice tubes located between the top and bottom tube sheets of the vessel. The fuel assemblies are loaded in these lattice tubes, with the heavy water moderator filling the inside of the reactor vessel. The reactor vessel is surrounded by a reflector consisting of two annular rings of graphite, cast iron thermal shields and a 2.5 meter thick heavy-concrete biological shield. On top and bottom of reactor vessel, there are aluminium and steel thermal shields cooled by demineralised light water with removable concrete biological shields placed at the top. (Figure 1).

The fuel is cooled by demineralised light water, flowing through the fuel assemblies from top to bottom, and circulating in a closed loop. Heat from the primary coolant is transferred to seawater in a shell and tube type heat exchangers with seawater flowing on the tube side in a once-through mode. Shut down cooling (decay heat removal) is provided by one-pass gravity assisted flow of water from a concrete storage tank (commonly known as Ball tank) of 3.8 megalitres capacity located at a higher elevation than that of the reactor and connected to the system through a set of check valves. Coolant outlet from the core is led to an underground tank through a set of quick opening valves located in the primary coolant outlet line (Figure 2).

3. Refurbishment and dismantling work

As a part of the refurbishment activity, several out of core components and structures including piping, equipment, supports etc. were dismantled and removed from the systems; some of these were repaired and reinstalled. Some were replaced by new ones. Execution of such work has given a hands-on experience in dismantling, handling and disposal of these materials from a decommissioning point of view. In the following sections, some of these are described.

3.1. Dismantling of coolant outlet cross header

At CIRUS, the individual fuel channel coolant outlets are connected to 17 cross headers with the cross headers in turn connected to an outlet ring header (Figure 3).

One observation during the refurbishing outage was the development of a crack adjacent to a weld in one of the coolant outlet cross headers made of SS 347 material. In situ metallographic examination was carried out by transferring the microstructure to a back reflecting plastic replicating strip. It was revealed that the area surrounding the crack had a sensitized microstructure. The crack propagation was seen to be inter-granular. Due to site constraints, detailed In situ investigation and repair work could not be carried out.

To carry out further investigation on the cross header, which has 13 welds, and to carry out the necessary repairs, the outlet header was dismantled. Dismantling work involved removal of a 100 mm plug valve at one end and cutting the other end of the cross header at the vertical leg.

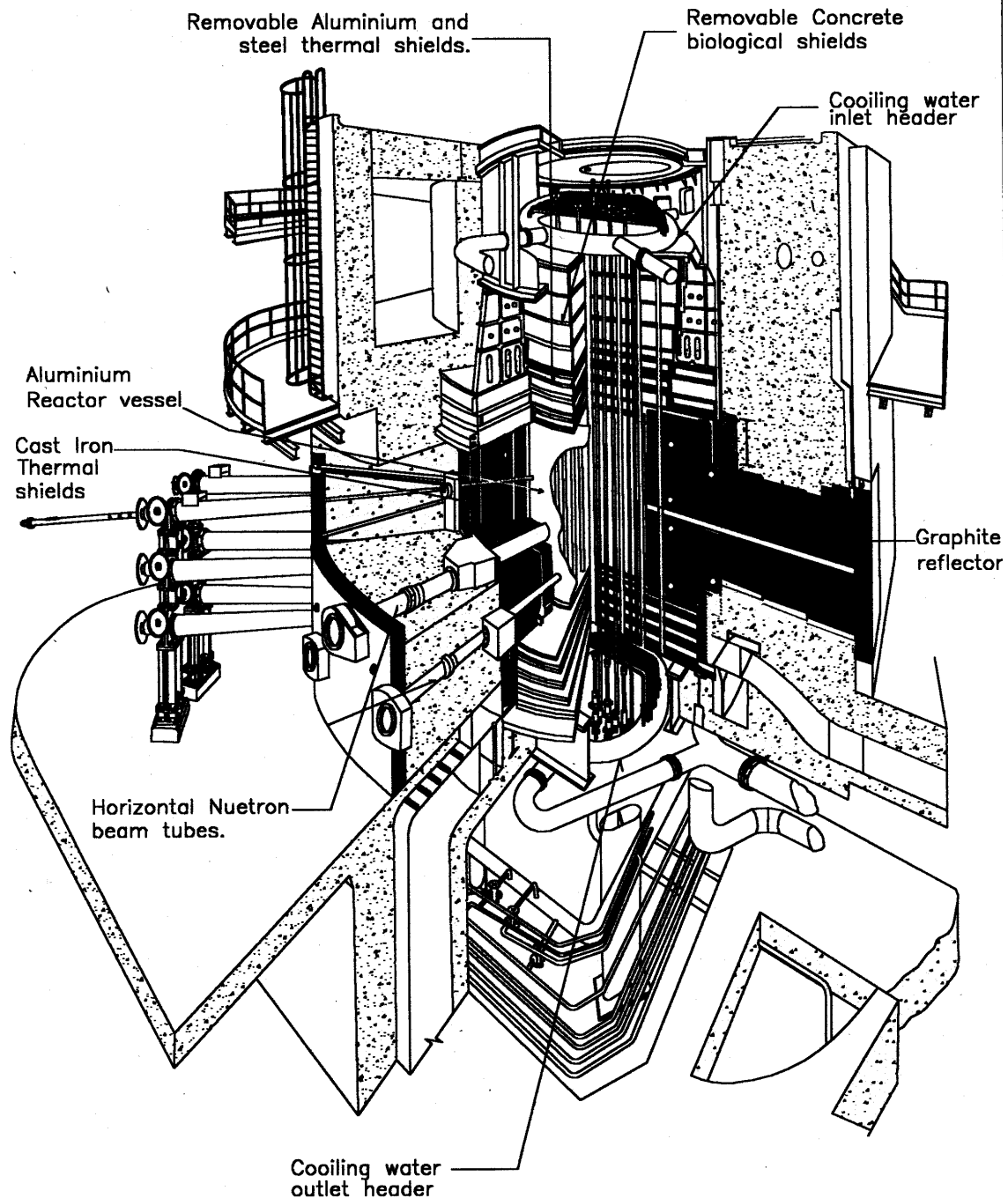


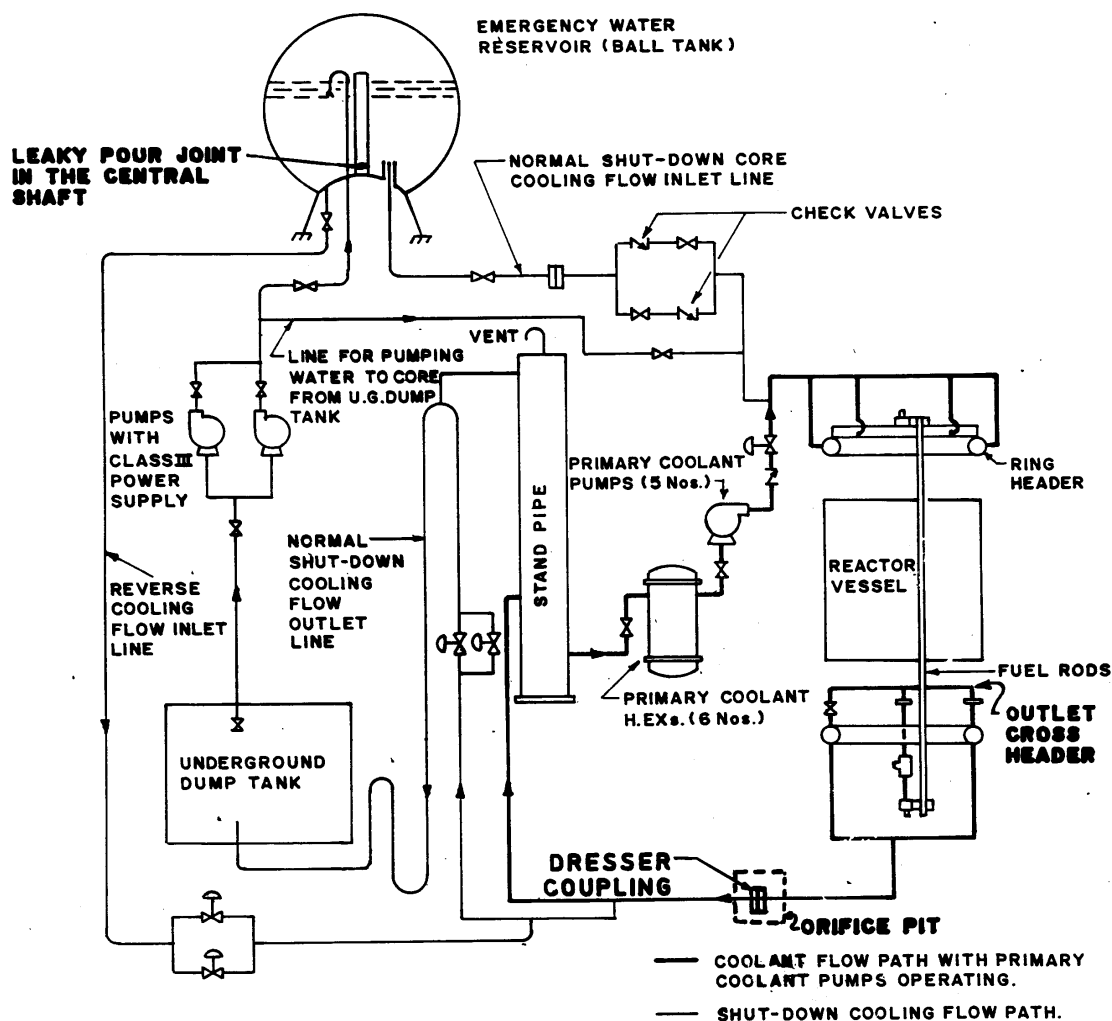
Figure 1. CIRUS reactor structure.

In addition, a number of stainless steel supporting members, which are welded with the cross headers to ensure fuel channel vertical alignment, had to be dismantled. Due to site constraints, standard pipe cutters could not be used for dismantling. Different cutting methods were evaluated and finally mechanical cutting using hand operated conventional tools and a pneumatically operated saw were found suitable. Before dismantling, reference markings were made with respect to adjacent cross headers to aid in the proper installation and alignment after repair and inspection.

After dismantlement, the cross header was decontaminated to bring down the radiation levels by a factor of 10 and after carrying out welding trials, the cross header was repaired by qualified welders using approved procedures. Subsequently the cross header has been installed back in position. This work provided a good hands-on experience in dismantling of an active component from a constrained area and its decontamination.

3.2. Repair work on subsoil pipelines

The major portion of the primary cooling system piping is of seamless carbon steel conforming to ASTM-A53 with diameters of pipes ranging from 50 mm to 500 mm. About 70% of the piping is laid in subsoil 5 m below ground level and individual sections of piping are joined with mechanical couplings having elastomer seals called "Dresser Couplings". In the 500 mm dia. lines, over 20 such couplings are provided.



**CIRUS-PRIMARY COOLING WATER SYSTEM
(SCHEMATIC)**

Figure 2.



Figure 3. View of coolant channel outlet cross headers.

To detect leakage from subsoil pipelines and to check the migration of radioactivity, a number of bore wells are provided in and around the reactor complex. Water from these bore wells and vegetation in the reactor complex is periodically sampled to check the radiological status of the environment around the reactor.

During the refurbishing outage, all primary coolant pipelines were tested at 110% of operating pressure to check their integrity. Most sections passed the inspections. These then underwent a metallurgical investigation to determine their fitness for the proposed life extension. Detailed checking of the sections that would not hold the test pressure revealed the leaky portions to be in subsoil region. Underground pipelines were exposed in some areas after excavation with due radiological precautions. Acoustic Emission Technology was used in an attempt to detect leakage but this did not work. Subsequently the leaky section was identified by the introduction of Fluorescein Sodium Dye in the coolant water and observing the subsoil water collected in dug out pits for evidence of the dye.

All subsoil piping (Figure 4) was exposed by excavation around and above these pipes. Various methods such as: visual inspection; pressure testing; ultrasonic testing; and testing of the protective coating were used for assessment of their condition. An estimated 8,000 m³ of soil was removed and about 1600 m length of primary coolant pipes were inspected. Based on the inspection, a plan was drawn up and carried out for the replacement of protective coatings, of elastomer gaskets of all couplings and of leaky pipelines. Due precautions were taken during the excavation to continuously monitor radiation fields and analyze soil and water samples for radioactive contamination. Contaminated soil was segregated from clean soil and disposed of following approved waste management practices. As expected, it was observed that the soil around the leaky zones of the pipes was contaminated.

Samples of pipe pieces were cut and radiological characterization was carried out to determine the extent and type of radioactive nuclides present. Radiation surveys also provided a reasonable idea of the extent and location of deposits. A few trials of mechanical cutting and also gas cutting were carried out to establish the methods. About 900 m of pipes in sizes ranging from 50 mm to 250 mm in diameter were thus cut and removed from the site and disposed of as radioactive waste.

The above work is expected to provide experience in dismantling of radioactive subsoil pipelines. This has also established that there has not been any major leak from subsoil primary coolant pipelines.

3.3. Replacement of Heat exchangers

Primary coolant heat exchangers are floating-head shell and tube type. The tubes and tube sheets are of 70:30 copper-nickel alloy and the corresponding channel covers are made of silicon bronze. Shells including bottom cover are made of copper bearing carbon steel. There are six heat exchangers of which five are in service and one is on standby. Heat exchangers are about 4 m in height and 1.2 m in diameter. The weight of tube bundle is about 3.5 tonnes and that of the complete assembly about 5 tonnes. Heat exchangers are mounted vertically on a steel support structure, which is grouted in the concrete floor to transfer the load of heat exchangers, pipes and valves in the area.



Figure 4. View of subsoil primary coolant pipes with fresh cold applied self adhesive bituminous tapes before backfilling.

As a part of ageing studies, the heat exchangers were inspected in detail. Inspection revealed severe erosion of tubes of the heat exchangers mainly in the section of the tube bundle facing the primary coolant inlet nozzle. Metallographical examination of damaged tubes indicated a non-uniform corrosion layer of thickness 20 to 170 microns and de-nickelification of the tube at the location of damage. The failure was attributed to progressive removal of the corrosion layer due to impingement of primary coolant water at the inlet. Also, channel covers on the seawater side had eroded substantially. The occasional spillage of seawater in the area and consequent seepage of water into the concrete and stagnancy of accumulated water around supports has led to external corrosion particularly at the bottom of supports.

During refurbishment, the Cupro-nickel tube bundles of the heat exchangers and steel support structures were dismantled and replaced with new ones.

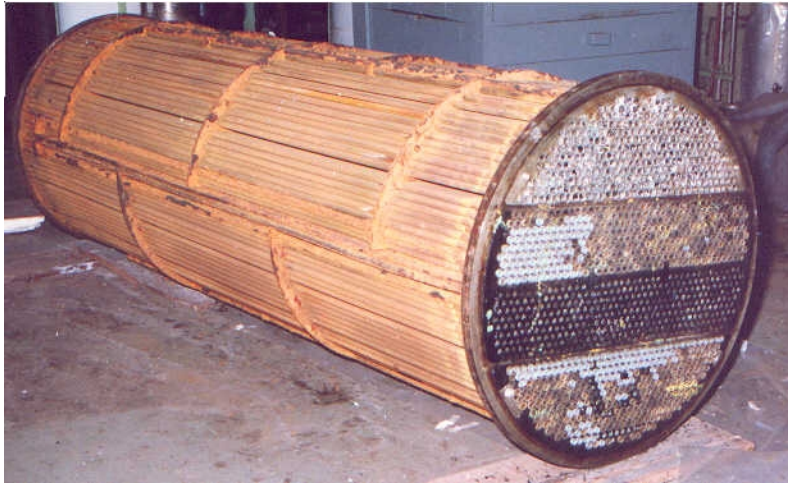


Figure 5. Primary coolant Heat Exchanger tube bundle in interim storage.

All the old tube bundles of these heat exchangers were decontaminated In situ to remove all loose contamination and transferred from the site to an interim storage area (Figure 5). The support structure was removed from site after chipping out the concrete floor. Support beams were removed as intact pieces without cutting.

The work has provided experience in dismantling of large radioactive components and their disposal methods.

3.4. Dismantling of primary coolant 500 mm headers

Three headers of the primary cooling system, 500 mm dia and 14 m long, are located in the Heat Exchanger room of CIRUS. As the environment in this area is saline and prone to seawater spillages, the rate of corrosion of components is higher. As a part of the refurbishment, these headers were inspected by visual and various NDT methods. Due to site constraints, it was necessary to remove the headers from their location and repair them based on the inspection. Due to corrosive environment several of the bolts of the header flanges were jammed in the bolt holes. Special wrenches, jacks and gas cutters were used to remove these bolts. Also, since the headers could not be removed due to interference from other pipes and supports, these supports had to be cut to remove the headers. This experience will be very valuable for dismantling and removal of heavy piping from a constrained area.

3.5. Replacement of moderator cover gas pipelines

The moderator and cover gas piping is made from SS 304 with diameters ranging from 6 mm to 200 mm. Some of these pipes have shown leaks due to chloride induced stress corrosion cracking of the SS components. It was observed that chlorides have deposited on the inside surface of the helium cover gas pipelines and, over the years, cracks have developed in certain sections. These pipe sections were cut and investigated. There was evidence of corrosion attack initiating from the inner side of the helium cover gas lines along chromium-depleted regions.

Several sections of these pipes, totalling 150 m in length, were cut by mechanical cutting means, removed and replaced. Dismantled pipe sections were disposed of as active waste after characterization of the contained radioactivity.

4. Assessment of (Wigner) stored energy in irradiated graphite reflector

CIRUS uses graphite as a reflector material. At the rated power of reactor operation, the graphite has been subjected to neutron irradiation for over three decades. This graphite reflector consists of two co-axial cylindrical structures around the reactor vessel (RV) and is cooled by ventilation air. Between the 228 mm thick inner cylindrical block (weight ~10 t) and the 622 mm thick outer cylindrical block (weight ~36 t), there is an air gap of 63 mm through which the reflector cooling air passes from top to bottom. (Figure 1)

For several years, prior to the refurbishment shut down in 1997, the CIRUS reactor was operated at 20 MW. It was considered that this could have increased the Wigner energy in the graphite due to decreased concurrent annealing during this period. Thus, before undertaking the refurbishment outage for the life extension of CIRUS, it was considered necessary to assess the amount of Wigner energy in the graphite and to assess the nature of the stored energy release curves to rule out any large spontaneous temperature rise due to heating of the graphite during reactor operation at the new rated power. An assessment of Wigner energy is also necessary for the safe disposal of irradiated graphite during decommissioning. This assessment would determine the treatment required to be given to graphite e.g. thermal annealing before disposal.

4.1. Measurement of Wigner energy release

Differential Scanning Calorimetry (DSC) was used for the measurement of Wigner Energy release. Sampling was carried out on a reflector graphite plug, 108mm (4¼ inches) diameter and 1.5 m (5 feet) long, that was extracted from the mid height location of the east thermal column. This plug (GR-I) was expected to represent the portion of the reflector, which is subjected to the maximum neutron fluence and consequently the worst conditions of stored energy.

Linear heating experiments were carried out to generate DSC plots on samples taken from different locations radially on graphite plug GR-I (Figure 6). The DSC plots gave the results of energy release in the units of mW (mcal/s). These values were re-plotted after converting to the units of specific heat, J/g°C (cal/g°C) to give the Wigner energy spectrum (Figure 7 & Figure 8). It can be seen from the figures that all the Wigner energy spectra, except the one for the sample from 305mm distance from the RV side end of the plug, are seen to lie below the specific heat curve. The Wigner energy spectrum of the sample from 305 mm location is seen to be crossing the specific heat curve at a temperature of 195°C. From equivalent area treatment, it was seen that this would lead to a sudden temperature increase to 260°C.

Isothermal annealing studies were also carried out on samples from graphite plug GR-I to work out a safe annealing procedure. The possibility of partial irradiation annealing of the Wigner energy in CIRUS reflector graphite was studied by carrying out isothermal annealing experiments above 120°C for an extended period.

Wigner energy release spectra and related studies have established that there is no possibility of uncontrolled Wigner energy release at any measured location of CIRUS graphite reflector. Also, the detailed DSC studies are expected to provide a good database for planning the safe decommissioning and disposal of irradiated graphite.

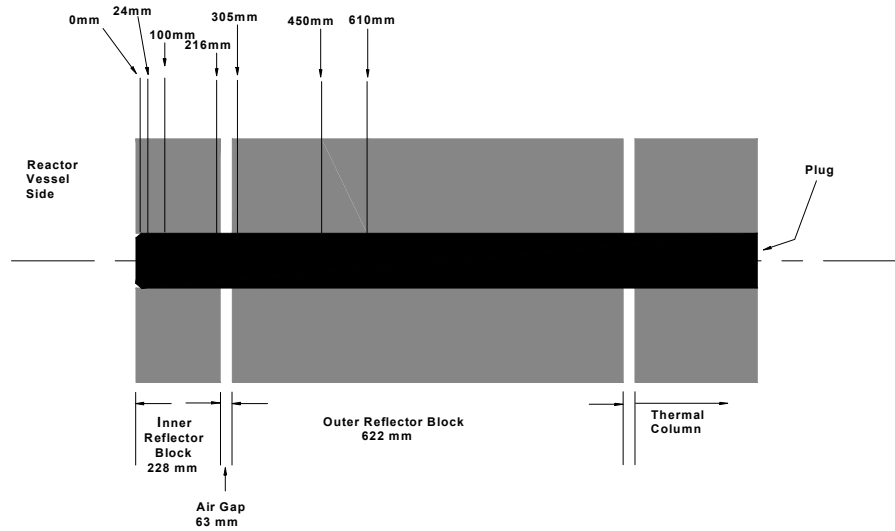


Figure 6. Sectional sketch of the graphite plug showing the sampling plan for Wigner energy release studies.

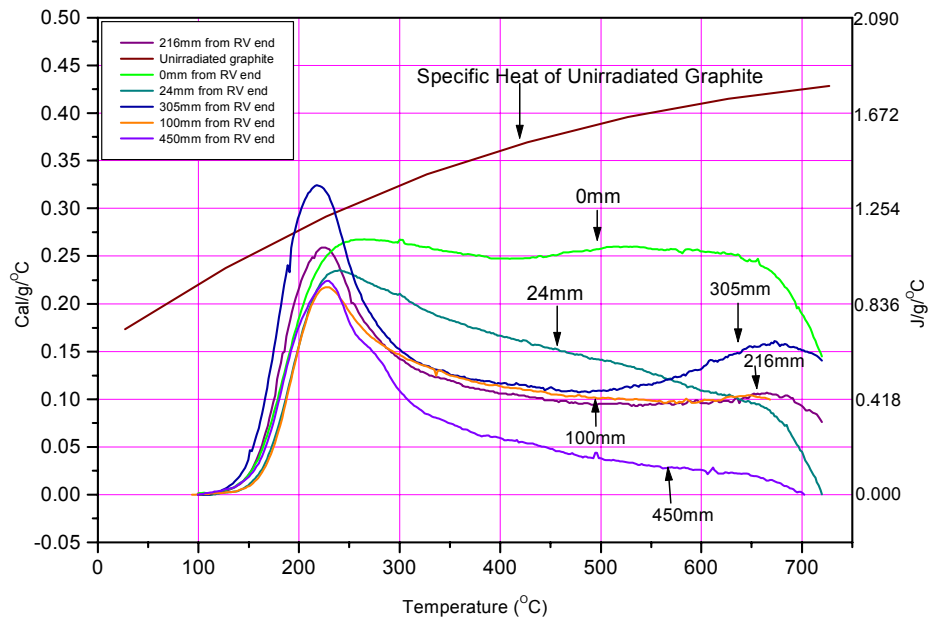


Figure 7. Wigner energy spectra of the samples from irradiated graphite plug (GR-1) of CIRUS after about 37 years of reactor operation.

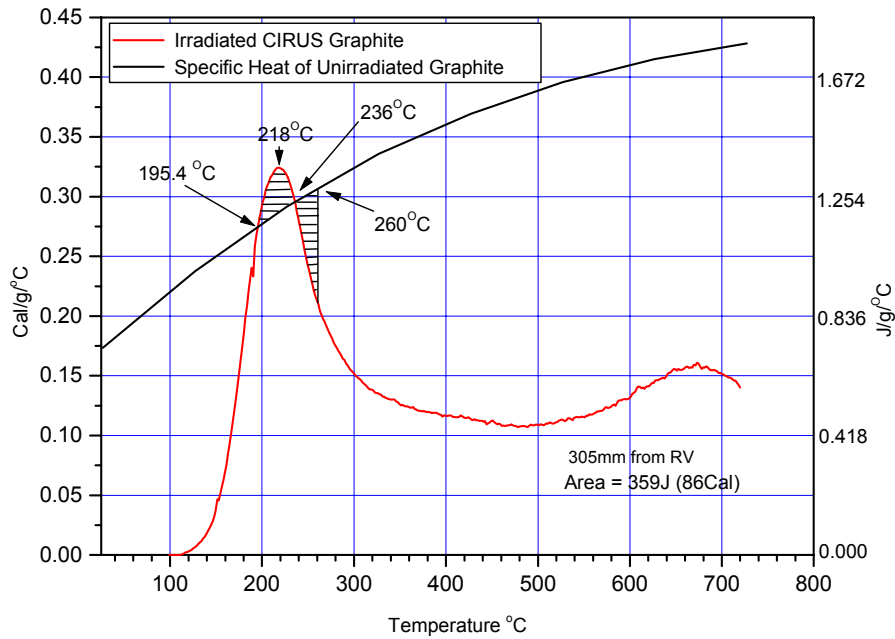


Figure 8. Wigner energy spectrum of sample at 305 mm from RV side end of irradiated graphite plug (GR-I) of CIRUS after 37 years of service showing spontaneous temperature rise during linear heating.

5. Radiological characterization and decontamination studies

During refurbishment several components of the reactor including piping, storage tanks, sumps and other equipments were accessible and available for extensive characterization, decontamination and radiation surveying. Samples from most of the components were analyzed and the results recorded. Most of the samples were analyzed using high resolution HPGe detectors. This is expected to give reasonable information on the extent of radioactivity present in various components to be decommissioned. Since the data has been collected after 37 years of service, a reasonable extrapolation would be possible with further information during in service radiation surveys. All the data was generated following a reasonable decay period after reactor shutdown which allowed the short lived radionuclides to decay. As can be seen from the data (Table I) ^{137}Cs is the dominant fission product and ^{60}Co among the activation products with other radio-nuclides like ^{90}Sr , ^{125}Sb , ^{144}Ce , ^{152}Eu , ^{65}Zn , ^{154}Eu , ^{95}Nb and $^{110\text{m}}\text{Ag}$ present in small quantities.

5.1. Radiological characterization of primary cooling system components

5.1.1. Expansion tank

A carbon-steel tank, 1.5 m dia. 17 m high, serves as expansion tank for the primary cooling system (Figure 9). During the refurbishment outage, the tank was drained, decontaminated and planned repair work was carried out on the corroded base plate following an approved procedure after conducting mock-up trials. After draining the expansion tank a radiation survey was carried out before beginning the repair work. The fields were ~ 0.5 mGy/h (50 mR/h). Crud in the form of slurry (about 900 litres), which had accumulated at the bottom of the tank, was removed as radioactive waste and decontamination of the inner surface and its base plate was carried out.



Figure 9. View of bottom part of expansion tank (Stand pipe).

Table I. Typical characterization data of different components

	Primary coolant Pipes	Fuel channel isolating valves	Primary coolant Heat Exchangers	Primary coolant expansion tank	Hot spots in Reactor structure cooling air ducts
Gross sp. act. (Bq/g)	6.6	5.6×10^4	1×10^2	1.5×10^3	9.1×10^5
Fission products	50 –90 %	47 %	55%	72%	
Activation products	10–50%	53%	45%	28%	>99%
Major nuclides contributing to gross activity	^{51}Cr ^{137}Cs ^{124}Sb	^{60}Co (42%) ^{137}Cs (22%)	^{60}Co (25%) ^{137}Cs (22%) ^{152}Eu (15%) ^{125}Sb (16%)	^{60}Co (26 %) ^{137}Cs (32%) ^{144}Ce (22%)	^{60}Co (99%)

After removal of the slurry and decontamination, the radiation fields came down to 0.1 mGy/h (10 mR/h). After welding of an additional new base plate with concrete filling over the old base plate, the radiation levels inside the expansion tank decreased to 10 $\mu\text{Gy/h}$ (1 mR/h). The collective dose for the repair job, including decontamination work, was 10 Person-mSv (1 man-rem).

5.1.2. Primary cooling system pipes

As mentioned in previous sections, samples of primary coolant pipes that were replaced were assessed for activity characterization. Sample pieces were cut from the core outlet pipes, specifically from the bends, etc., where deposits are expected to accumulate. Samples were cut using mechanical tools and flame cutting was avoided. Scrape samples from the inside surfaces of the cut pieces of pipes were collected until the substrate was exposed. These samples were then analyzed using a high-resolution HPGe detector. It was seen that average specific activity was only 6.6 Bq/g and the bulk of it belonged to ^{137}Cs (Table I). Radiation fields on the pipes were within the range from 2 to 7 $\mu\text{Gy/h}$ (0.2 to 0.7 mR/h)

5.1.3. Primary cooling system heat exchangers

These shell and tube type heat exchangers have the primary coolant on the shell side. It is therefore expected that contamination will be present on inside of the shell and the outside of the tubes.

Scrape samples were taken from the inside of the shell and analyzed. Typical results are shown in Table I. Data revealed that deposits on the shell inner surface were higher at lower elevations of the heat exchangers.

Some of the Cupronickel tubes removed from Heat Exchanger tube bundles were characterized for the presence of radionuclides. Water, with normal detergents, was used to remove loose contamination prior to characterization studies. It was seen that the ^{137}Cs was the main constituent with small traces of ^{125}Sb and ^{60}Co .

5.1.4. Fuel channel isolating valves

The primary coolant water is fed to individual fuel rods through a set of 17 cross headers branching off from the main supply header (Upper Ring Header). Similarly water coming out from individual fuel rods is led to the main outlet ring header through cross headers. To facilitate fuel replacement in under reactor shut down conditions, isolating valves are provided both in the upper and lower cross headers. These valves, made of aluminum-bronze alloy, were found to be pitted due to corrosion/erosion. During the current refurbishing outage, these valves have been replaced by SS 316 investment cast valves.

Radiation field on the valves varied from 0.01 to 0.30 mGy/h (1 to 30 mR/h). The high radiation field observed on some of the valves was due to the deposition of activation and fission products over a long period of operation. Quantitative and qualitative analysis of deposits on the valves were carried out to estimate the quantity of activity present.

Internal deposit samples from the valves at the outlet, with radiation fields of ~ 0.05 mGy/h (~ 5 mR/h), were collected over an average area of 100 cm^2 . The samples were collected after two years of reactor shut down to facilitate the decay of short lived components. The corrosion film thickness was assumed to be uniform all over the exposed area. From the specific activity (Table I), the average total surface activity of each valve was found to be 280 Bq/cm^2 . The total activity on the 182 outlet isolating valves is thus estimated to be 11.52 MBq.

5.2. Characterization of soil around the reactor complex

As mentioned above, several of the primary coolant and waste transfer pipes are laid under ground but separated from the various utility systems within the plant boundary. To detect leakage from subsoil pipelines and to check migration of radioactivity, a number of

bore wells are provided in and around the reactor complex. Water from these bore wells is periodically sampled to check the radiological status of the environment around the reactor.

Also, during the excavation of these pipes, several soil samples were collected in and around the plant boundary at varying depths. Most did not show any activity; however, at some places soil was seen to have radioactivity at depths from 1 m to 5m below ground. This was attributed to some leaks from the pipelines during the initial days of operation of the reactor. These pipes have since been taken out of service. As a part of their surveillance, all subsoil pipes are pressure tested at periodic intervals to test for leaks.

The soil samples collected have clearly identified the areas where activity has been trapped in soil. This information will be very useful in monitoring the area through bore well samples and for eventual clean up operation during decommissioning.

It was seen that ^{137}Cs is the dominant radio-nuclide activity ranging from 56 Bq/g to 1600 Bq/g with traces of ^{134}Cs , ^{152}Eu and ^{154}Eu .

5.3. Decontamination studies

5.3.1. Heat exchanger tubes

Radionuclides along with other corrosion products, which are circulating in the primary cooling system, get deposited on the surface of tubes. It is therefore apparent that full-scale decontamination is possible only when the corrosion layer is dislodged. It was seen that corrosion products deposited were mainly hematite (Fe_2O_3). High-pressure water jets with pressures up to 1000 bars have been found quite effective. Chemical decontamination with strong acids is also seen to be quite effective.

These tubes constitute a significant quantity (21 t of cupronickel material for all six heat exchangers) in terms of waste to be disposed.

The two methods discussed below were tried.

In Method I a one step reducing formulation containing 2% w/w Na-EDTA, 5% w/w ascorbic acid and 1% w/w hydrazine hydrate was used. A decontamination factor (DF) of 1.5 could be achieved by this method.

In Method II a solution of 4% w/w HCl with 0.5% w/w ascorbic acid as inhibitor was used at room temperature to decontaminate the samples. It was seen that within a 24 hour period, practically all the radioactivity and corrosion products had come into solution, decontaminating the samples completely.

5.3.2. Fuel channel isolating valves

Experiments were conducted using conventional decontamination reagents like EDTA, oxalic acid, and citric acid to decontaminate the valves. As the valves were removed for decontamination two 2 years after reactor shutdown, most of the short lived activity had already decayed. The major contribution to the significant radioactivity on these valves came from ^{60}Co and ^{137}Cs . From the experiments conducted, it is seen that a preoxidizing treatment followed by a reducing treatment gives a good decontamination factor of around 20. It is observed that EDTA is a necessary reagent in the formulation to achieve a good decontamination factor. Low concentration of the reagents can be employed, if the temperature of the solution is raised to a higher value. The duration of the treatment can be extended to achieve a higher decontamination factor. The pH value of the solution was

appropriately chosen so as to have less corrosion of construction materials, thus minimizing the consumption of ion exchange resins used for trapping radioactivity.

The process can be further improved to optimize concentration, temperature and time to achieve better decontamination factors.

Gamma spectrometric analysis of samples of the decontaminant solutions after decontamination was carried out. Results are as shown in Table II below. It shows that a DF of 20 could be achieved.

Table II. Data for decontamination of fuel channel isolating valves

Construction material exposed	Decontaminant Formulation	DF Achieved	Radio nuclides detected	Radioactivity Removed Bq (μCi)
Stainless steel	EDTA +Oxalic acid	5	⁶⁰ Co ¹³⁷ Cs	2.1×10^5 (5.7)
Aluminium bronze	Pre-oxidation treatment followed by EDTA + Oxalic acid + Citric acid	20	⁶⁰ Co ¹³⁷ Cs	6.66×10^4 (1.8)

5.4. Decontamination for inspection and repairs

5.4.1. Reactor ventilation exhaust duct

The reactor ventilation exhaust duct runs between the reactor structure cooling outlet and the inlet plenum of the high efficiency particulate air (HEPA) filter banks. This is a 1500 mm dia. carbon steel duct embedded in concrete running over a length of 28 metres. During the current outage, this duct was inspected in detail and was decontaminated to carry out necessary repairs. The duct was found to have several hot spots of the order of 20 mGy/h (2 R/h) and was also found to be corroded internally at certain locations.

Prior to entering the duct, detailed planning was done after studying the drawings. The necessary radiological and industrial safety precautions were taken. Necessary lighting arrangements were made and the visual inspection was video taped. The hot spots were decontaminated after scooping out the radioactive sludge with specially fabricated tools. The total dose consumed for the entire operation of cleaning and videography was 10 Person-mSv (1 man-rem). Gamma spectrum analysis of the radioactive sludge indicated presence of mainly ⁶⁰Co and small quantity of ¹³⁷Cs. The maximum specific activity found was 9.1×10^5 Bq/g.

5.4.2. Wet Storage Block (WSB) – an interim fuel storage facility

The Wet Storage Block (WSB) is a facility for storing around 200 spent fuel rods within the reactor containment building before they are transported to the Spent Fuel Storage Building (SFSB) for further processing. This is a rectangular concrete block, 2 m x 1.6 m x 10 m deep, lined with carbon steel. Water level is maintained at 9.5 m at all times to ensure full submergence of the stored fuel and safety provisions exist to take care of any inadvertent draining. The water is kept in circulation to maintain water chemistry.

After the reactor shut down in Oct. 1997, the complete core was unloaded and all the spent fuel was stored in the WSB for the required cooling before further transfer and processing. During May 1999, the spent fuel rods were transferred to the SFSB. Subsequently, to carry out an inspection, the water in the WSB was drained to a waste disposal tank using a submersible pump. The crud at the bottom was pumped as a slurry into a waste disposal drum with due radiological surveillance. The WSB was decontaminated by flushing with water jets to reduce the radiation level to allow a detailed inspection of the carbon steel liner and to make repairs as required.

Prior to decontamination of the WSB, the radiation fields on the sidewalls were in the range of 0.1 to 1 mGy/h (10 to 100 mR/h) and at the bottom, 2 to 20 mGy/h (200 to 2000 mR/h). With decontamination, the radiation fields were lowered by a factor of 5.

Before decontamination samples, taken from the WSB sidewalls were characterized. The results indicated that the specific activity in the lower half of the WSB was higher than in the top half. The predominant radio nuclides present were: ^{137}Cs , ^{125}Sb , ^{144}Ce , ^{152}Eu , ^{65}Zn , ^{60}Co , ^{154}Eu and $^{110\text{m}}\text{Ag}$ (Table III).

5.4.3. Shut down cooling system — concrete storage tanks

The spherical concrete water storage tank (Ball Tank) of the shut down cooling system (Figure 10), located at a higher elevation than that of the reactor, had developed a small leak some years back at a concrete pour joint in the central inspection shaft. During the refurbishment outage, the water in this concrete tank was completely drained and repairs made. The tank had some low level contamination on its inner surface. A detailed radiation survey and characterization was carried out after decontamination prior to beginning the repair work. The sludge collected during decontamination had a specific activity of 10 Bq/g (Table III) amounting to total radioactivity of 0.04 MBq. Radiation fields on the decontaminated concrete surface of ball tank were 1 to 2 $\mu\text{Gy/h}$ (0.1 to 0.2 mR/h). The maximum radiation field was 20 $\mu\text{Gy/h}$ (2 mR/h) at the crack region at the base of central inspection shaft.

The one-pass gravity assisted shut down cooling water from the reactor is led to the cubical concrete water storage tank (Dump Tank) located underground. It receives water from the ball tank after cooling the fuel assemblies during reactor shut down. The tank has 3 sections; Section 1 & 2 for collecting highly radioactive water (during a failed fuel condition) and section 3 for collecting water during normal condition.

Table III. Typical data from different tanks and sumps containing radioactive fluids slurry samples collected during clean up operation

	Dump tank	Ball tank	Wet storage block	SFSB area sump	Non chemical main sump	Chemical main sump	System water - Catch tank	Floor drain tank
Gross sp. Act.(Bq/g)	48	10	2298	1.1×10^5	1.1×10^4	8.3×10^4	238	1.1×10^5
Fission products	>99%	>99%	>98%	>99%	>99%	>99%	92%	>99%
Activation products							8%	
Major radio-nuclides	^{137}Cs	^{137}Cs	^{137}Cs	^{137}Cs	^{137}Cs	^{137}Cs	^{137}Cs (91%)	^{137}Cs

During the refurbishment outage, this concrete tank was also completely emptied, cleaned and inspected. The tank sections were cleaned using high-pressure water jets (200 Kg/cm^2). Sediment samples were taken from the bottom of the tank and characterization studies were conducted (Table III).

A detailed radiation survey was carried out, after decontamination, prior to the inspection. Radiation fields on the decontaminated concrete surfaces of the dump tank were $< 1 \text{ } \mu\text{Gy/h}$ ($< 0.1 \text{ mR/h}$). It was observed that the concrete tanks could easily be decontaminated to bring the radiation levels down to low values. The epoxy painted surface was found to be in good condition.

5.4.4. Liquid effluent sumps

CIRUS liquid waste disposal system incorporates eight sumps situated in different areas of the reactor and spent fuel storage buildings. The sumps are classified as radioactive or inactive and also as chemical or non-chemical. The sumps are concrete chambers below ground level and have a lining of mild steel for non-chemical sumps and stainless steel for chemical sumps.

Liquids collected in these sumps are transferred to two outside active sumps in the liquid disposal area. The active liquids, after neutralization, are transferred to an effluent treatment plant for further treatment and disposal.

Forty years of operation has led to the collection of radioactive laden sludge deposits in all of these sumps. Characterization data of the sludge collected from some of these sumps is presented in Table III.

During reactor operation, it is difficult to work on the sumps, as a long outage of the sumps is not available, hence during the refurbishing outage all sumps were thoroughly cleaned/decontaminated, inspected and repaired. Further, as a part of the ageing studies, the integrity of the sumps was checked periodically to ensure no undue migration of radioactive liquids to surrounding areas takes place.

Clean up

Sumps pose a different challenge for cleaning.

- Entire clean up operation has to be done In situ unlike other components/equipments, which can be transferred to specialized decontamination centers.
- Unlike piping systems, it is not possible to have a flow of decontaminants through them, which is then collected.
- Unlike overhead tanks where gravity flow can be utilized, sumps are generally located below ground level, necessitating mechanical equipment such as pumps etc.
-

A typical clean up operation of a sump in spent fuel storage building is described below.

Spent fuel storage bay sump

This sump contained a collection of used tools, components and sludge in a fine slurry form. Due to the high radiation field at the bottom of the sump, 0.2 Gy/h (20 R/h), it was decided that the entire cleaning would be done with remote operated tools.

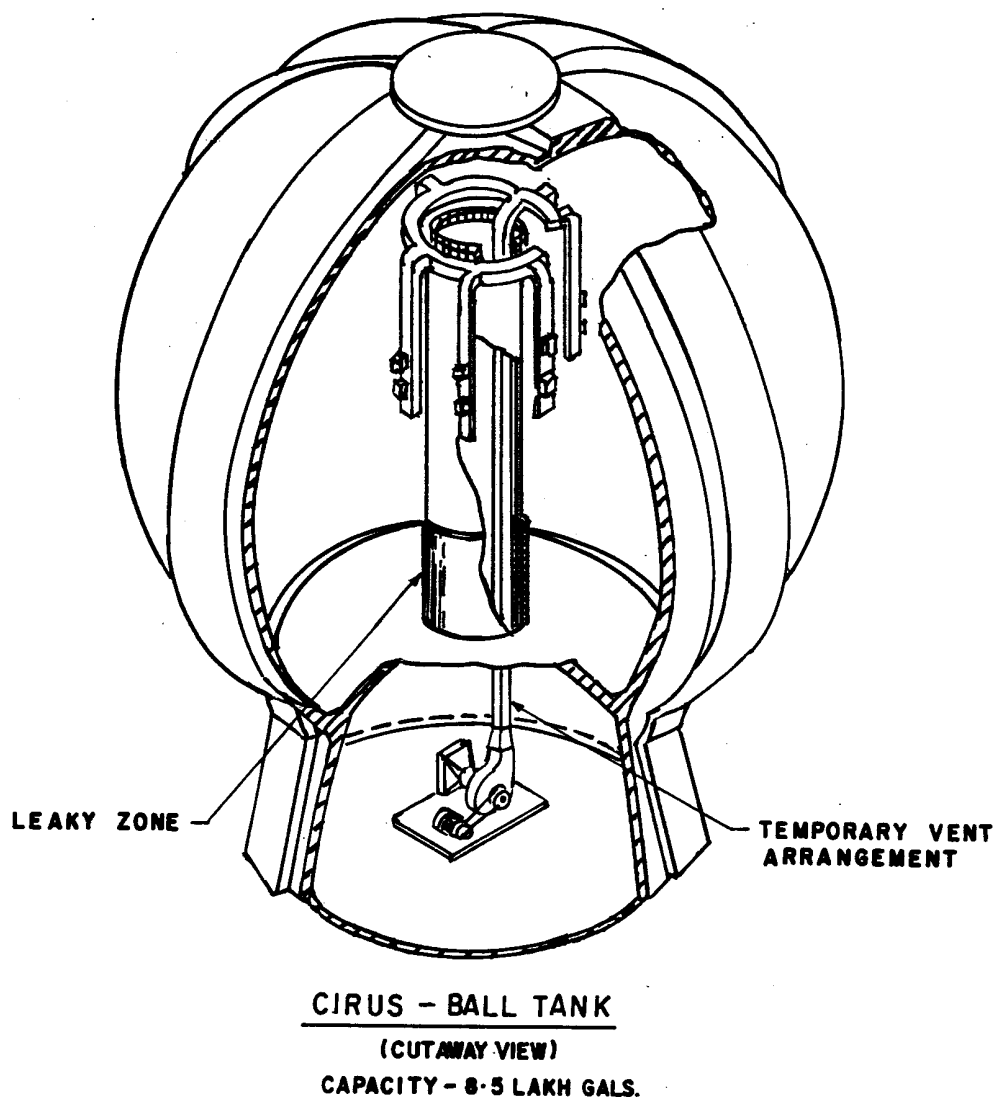


Figure 10. CIRUS ball tank.

A cage was suspended into the sump from above and heavier objects were transferred into the cage using long tools. The cage was then lifted out and washed to allow loose sludge to fall back into the sump. The objects retrieved were subsequently decontaminated.

The sump was dewatered using its regular pump, which left about one foot of water in the sump. A high-pressure jet pump with a special unit attached to it was used as a sludge suction pump to remove the water completely.

The sludge suction unit was also used to remove the sludge but was not successful for the following reasons:

- It could not lift the sludge beyond 3 m.
- Rate of removal of sludge was slow, which in a high radiation environment would result in higher man-rem consumption.
- Removed sludge along with the water had to be transferred to a settling tank.
- Sludge was fine in nature and could not be collected by filtration.

Hence a special scrapper and a folding type scoop were fabricated. This was used for loosening and scooping the sludge into a bucket. It was designed to make it lightweight and simple to operate from a distance. This was found very effective and about 75 kgs of sludge was removed from the sump. A lead lined tank was used for temporary storage of the sludge, before the final transfer for radioactive waste disposal. Subsequently the stainless steel lined sump was decontaminated using chemicals in two stages: first a formulation of EDTA and later with 1% HNO₃. This gave a DF of about 3.

Sludge was cleaned from all of the sumps in the CIRUS complex. A high-pressure water jet (200 bar) was used for cleaning the floors and lining. This was found very effective. Repair works on the lining, wherever required, were then carried out and the sumps were put back into operation. Experience shows that it was possible to clean the sumps with normal commercial decontaminants viz. detergents, citric acid, EDTA etc. after removing the sludge from the sumps.

6. Radiation survey of CIRUS reactor structural components

6.1. Estimation

An assessment of radiation fields from neutron irradiated structural components of CIRUS at the end of three decades of high power operation was carried out during 1995 based on neutron activation of the impurity elements present in the materials of construction. A 25-day operation — 5 day shut down cycle and four neutron energy group flux analyses was used in the study.

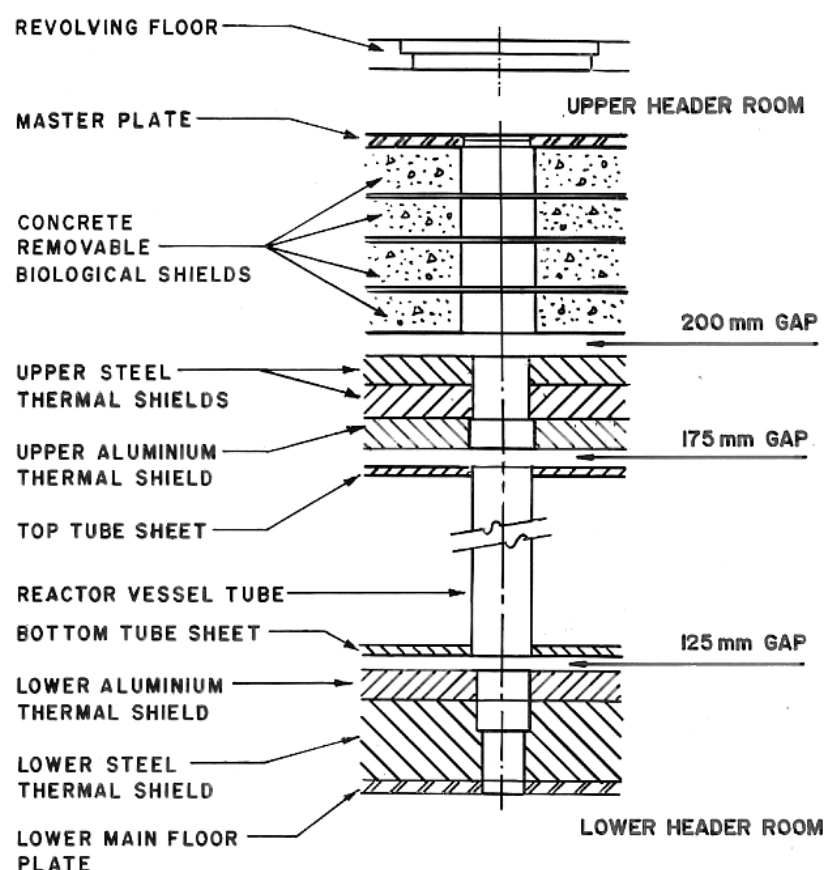
The individual surface gamma radiation fields in the reactor structural components after three decades of irradiation and after one year radiation cooling were estimated as follows:

- (a) 0.60 mGy/h -0.35 Gy/h (60 mR/h - 35 R/h) in the structural carbon steel thermal shield components above and below the reactor structure. The dominant radionuclides contributing to the surface gamma field are ⁵⁴Mn (>90%) half life = 312.2 d, ⁵⁹Fe (1-10%) half life = 44.6 d and ⁶⁰Co (<1%) half life = 5.27 y
- (b) 2.72–9.93 Gy/h (272 - 993 R/h) in the structural aluminum components above and below the reactor vessel. The dominant radionuclides contributing to the surface gamma field are ⁶⁰Co (>90%) half life = 5.27 y and the decay product ¹¹⁷In (<1%) half life = 1.93 h
- (c) 2.53-36.2 Gy/h (253 - 3620R/h) in the structural aluminum components of the reactor vessel. The dominant radionuclides contributing to the surface gamma field are ⁶⁰Co (>90%) half life = 5.27 y and the decay product ¹¹⁷In (<1%) half life = 1.93 h

6.2. In situ measurements in reactor structure components

Radiation field measurements on the reactor in-core components were carried out during the period from November 1997 to January 1998, following two months of reactor shut down. After completely unloading the reactor core, the radiation field measurements were determined in 46 pile positions at 20 different elevations between the upper header room (UHR) and the lower header room (LHR) (Figure 11).

The measurements were repeated during May 2001, following 43 months of reactor shut down, in 195 pile positions at different elevations.



**CROSS SECTION OF REACTOR
THROUGH PILE POSITION**

Figure 11. Vertical cross-section showing different components of reactor structure.

For the purpose of radiation surveying, a pre-calibrated modified version of a high range gamma monitor was used. It could measure a radiation field up to 10 Gy/h (0–1000 R/h). A modified version of the high range gamma monitor consists of a detector housed and sealed in a 30 mm dia and 100 mm long tube attached to a 250 mm long aluminium tube with a 0.75 Kg. dead weight. The detector system was connected to the monitoring system by a 20 m long flexible cable. The dead weight ensured that the detector and cable remain vertical during measurements in the lattice tubes. The readings were in good agreement with Thermo Luminescent Dosimeter measurements at selected locations.

6.2.1. Salient observations

Table IV presents a summary of the salient measurements made at two times following the shutdown, 2 months and 43 months. The highlights are :

- The minimum radiation fields seen were in the master plate and the removable biological concrete shields regions. These radiation fields were due to contamination. This was confirmed by gamma spectrometric analysis of swipe samples collected from the master plate, which showed the presence of long lived fission products such as ^{137}Cs . Decontamination of this plate brought the radiation fields down.

- The maximum radiation fields were observed just above and below the reactor vessel tube sheets. These are due to activated stainless steel cooling water tubing connected to the reactor vessel top and bottom tube sheets. A reduction of 2.5 to 5 was seen in the radiation fields in this region.
- In the center of the aluminium reactor vessel, a reduction by a factor of 2.5 was observed.
- In the top thermal shields, radiation fields are seen to be higher than earlier measurements. This is due to the fact that during earlier measurements, a few dummy assemblies were installed in some water-filled positions and therefore radiation fields observed were lower than with empty positions.

TABLE IV. RADIATION FIELDS IN REACTOR STRUCTURAL COMPONENTS

Reactor Components	Measured Radiation Fields Following Reactor Shutdown (SD)	
	2 months after SD	43 months after SD
Above reactor	10^{-2} mGy/h	
Master Plate	3–15	2 – 7
Removable Biological Concrete Shields	1–5	2 – 7
200mm Gap above Upper Steel Thermal Shield	20–50	15 – 50
Core area	10^{-2} Gy/h	
Upper Steel Thermal Shield	0.5–2	0.5 – 1.5
Upper Aluminum Thermal Shield	60–150	60 – 200
175 mm Gap above Reactor Vessel	200–500	50 – 350
Reactor Vessel	150–200	60 – 80
125 mm Gap below Reactor Vessel	200–800	40 – 200
Lower Aluminum Thermal Shield	60–200	20 – 80
Below core	10^{-2} mGy/h	
Lower Steel Thermal Shield	30–50	30 – 50
Master plate	20–30	10 – 20

6.3. Horizontal neutron beam tubes

In situ radiation field measurements on the eleven horizontal beam tubes (1 beam tube of 300 mm dia. and 10 beam tubes of 100 mm dia.) were carried out in March'98. For each beam tube, radiation field measurements were recorded at 21 radial locations in the concrete biological shield, cast iron thermal shields, graphite reflector and reactor vessel surface (Figure 1).

6.3.1. Salient observations:

Typical radiation fields for the 300 mm dia. and 100 mm dia. beam tubes are as indicated in Figure 12 below. The highlights are

- Measurements show that overall the fields were higher for the 300 mm dia. beam tube than for the 100 mm dia. beam tube. The highest field observed was 3 Gy/h (300 R/h) for the 300 mm dia. beam tube in the cast iron thermal shield region.
- For all the eleven beam tubes, the radiation field at the reactor vessel surface was 1 Gy/h (100 R/h).

- For the 300 mm dia. beam tube, radiation fields were from 3 to 200 mGy/h (300 mR/h to 20R/h) in the biological shield region; 0.6 to 3 Gy/h (60 to 300 R/h) in cast iron thermal shield region; and 600 to 800 mGy/h (60 to 80R/h) in the graphite reflector region.
- For the 100 mm dia. beam tubes, radiation fields were 0.25 to 6 mGy/h (25 to 600 mR/h) in the biological shield region, 25 to 700 mGy/h (2.5 to 70 R/h) in the cast iron thermal shield region and 350 to 700 mGy/h (35 to 70R/h) in the graphite reflector region.
- Two graphite plugs, which were removed from the east and west thermal columns in Jan.1997 and June 1998 for Wigner stored energy measurements, showed maximum radiation fields of 70 mGy/h (7 R/h) on contact.

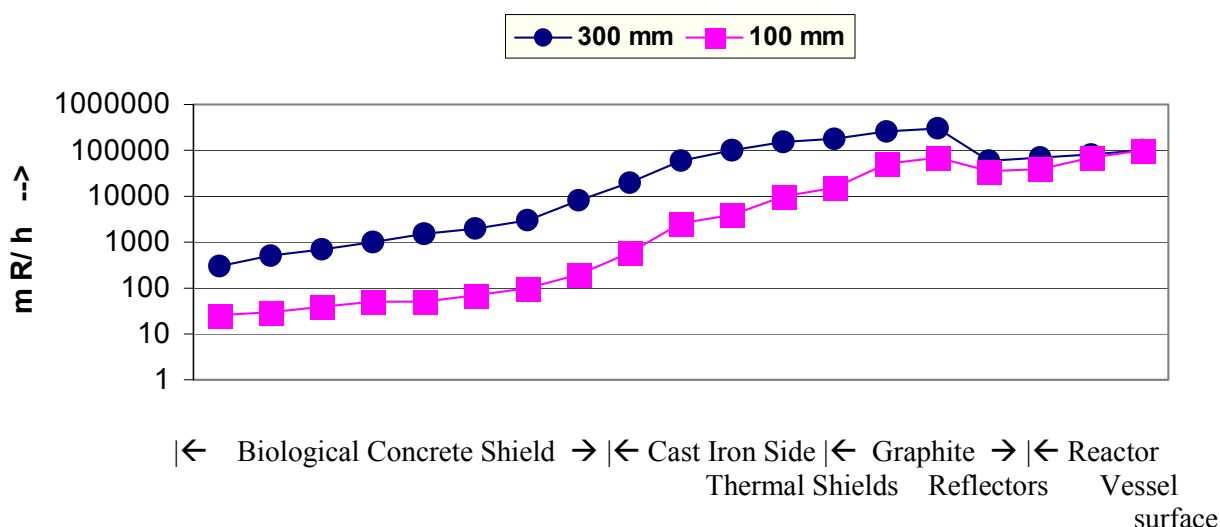


Figure 12. Radiation fields in horizontal neutron beam tubes.

7. Development of special equipment

7.1. Spent fuel storage bay cleaning equipment

The Spent Fuel Storage Building (SFSB) of CIRUS is located adjacent to CIRUS. Following interim storage in the reactor building, spent fuel is transferred to the SFSB by an underwater fuel transfer buggy through a canal filled with water. Bay floors often get filled with miscellaneous debris due to various fuel handling operations. Materials found include fine powder from underwater cutting of fuel shielding sections, dirt, uranium powder due to clad failed fuels etc. Periodically this radioactive fine debris needs to be removed.

A bay cleaning set up consisting of a cyclone separator, settler, filtration unit and a multistage pump with associated piping was designed, fabricated and operated in the Spent Fuel Storage Building. Its design features include the capability of:

- lifting heavy metals, including uranium from the bay floor;
- depositing lifted material into a disposable leak tight container;
- using bay water for transportation of debris in a closed loop; and
- minimizing airborne activity due to the disturbance of bay water.

7.2. Remote operation of 30 T E.O.T crane

CIRUS reactor structure (pile block) is situated inside a steel containment building of cylindrical structure with hemispherical top.

A 30 Tonne polar crane serves all material handling requirements. This crane was used for erecting the reactor structure.

The same crane is also required for dismantling the core components viz. various biological and thermal shields, reactor vessel and other structural components during decommissioning. An operator operates the crane from a cabin on top of the crane. This arrangement is used for routine work. During dismantling of the reactor structure, high radiation fields are encountered due to activation products in the core components. In order to prevent the crane operator being exposed to the high radiation fields, a scheme for remote operation of the crane was designed; constructed and tested.

The scheme envisages a number of mobile operating consoles connected to the control circuit of crane through multi pin quick connect control cables. The crane can be operated from these panels in conjunction with closed circuit TV camera units. Consoles will be located at selected locations depending on the expected radiation fields to be encountered. For normal operation, the crane may be operated from the cabin. As the radiation fields increase during dismantling operation, consoles kept away but in the direct line of sight may be used. However when the radiation levels are quite high, consoles would be operated from shielded areas using CCTV.

The consoles have been designed with the necessary interlocks to prevent any inadvertent operation of crane from more than one location. Also controls in the operator's cabin can be mechanically locked out when a remote console is in use.

The wiring has been done and the one available console was used to commission the system. It is expected that this will be very useful during dismantling operations when decommissioning of the reactor structure is carried out.

7.3. Remotely operated graphite sample cutting device

As described in Section 4.1 above, it is necessary to collect samples of irradiated graphite periodically for actual measurements of the Wigner energy to assess thermal safety aspects. This will also be required for decommissioning planning. For this, a remote, manually operated, graphite sample-cutting device was designed and fabricated. The device can be inserted in the 65 mm gap between the inner and outer reflectors and can cut samples from either side at any elevation. The driving unit is mounted above the reactor structure at a height of about 8 m from the sampling location.

The cutting tool is rotated through a set of gears by operating a hand wheel at the driving unit. Motion is transferred through a long vertical tube. An outer tube supports the tool head. A back support is provided to prevent swinging of tool head. As the cutting tool advances in the graphite, it will cut a cone of graphite 10.5 mm in diameter at its base and 9 mm high. A button sample 10mm in diameter and 2 mm thick can be machined out of this. The device has provisions for tool position indication; feed mechanism, a collection pot for collecting samples etc. This tool is in the final stages of development and is undergoing trials.

8. Achievements and lessons learned

Extensive radiological characterization of various systems, structures and components has been carried out. The data generated has provided very useful information on the type and extent of the radionuclides present. Available information can be integrated to determine the total inventory of radionuclides in the different systems, structures and components.

In situ radiation surveys of the reactor core and other structural components provides the necessary information for decommissioning planning and man-rem budgeting.

Experience with dismantling and partial decommissioning of several out-of-core components of the reactor indicates that these are not expected to pose much difficulty in future decommissioning. Dismantling by mechanical cutting tools is safer and without the problems of high airborne activities but it is slower and time consuming. More mechanized and automatic tools viz. automatic pipe cutters, will reduce the time and effort. Thermal cutting methods are faster and therefore should be employed in areas that are contained and ventilated through HEPA filters.

Studies carried out with various decontamination methods and the data generated will be useful in selecting the proper decontamination methods to minimize the radiation levels and contamination of decommissioning materials. Studies also suggest that decontamination using strong chemicals gives higher DFs and therefore is more suitable for decommissioning as compared to decontamination during operation. However, higher corrosion rates during decontamination increase the handling of consequent secondary liquid wastes and an optimization is therefore necessary.

Clean up operations on various radioactive sumps and storage tanks have given insight into the associated difficulties. The methods used need to be further improved.

Concrete decontamination is difficult and proper techniques to segregate the contaminated layer from rest needs to be further developed.

9. Future work for decommissioning planning for CIRUS

In accordance with the existing regulatory requirements, a decommissioning plan needs to be prepared and submitted to the safety authorities at the time of final shut down, which includes reactor details, decommissioning strategy, decommissioning organization and project management, decommissioning activities, quality assurance, safety analysis, radiological protection programme based on ALARA, radioactive waste management and a planned final radiation survey. This plan needs to be approved by safety authorities prior to the start of actual decommissioning.

9.1. Use of data

The data generated and the experience gained during refurbishment will be highly useful towards formulating the plan for the future decommissioning of CIRUS.

The characterization data collected can be used to arrive at the total inventories of radionuclides in the different systems, components and structures. The decontamination methods explored provide a basis for the selection of suitable decontamination processes. This will also help in identifying components that can be economically decontaminated for unrestricted release.

The In situ radiation surveys of CIRUS can be utilized to arrive at a composite half-life of radioactivity decay and therefore more accurately predict the radiation levels after specified periods of shutdown. This will be very useful in optimizing any deferral period of decommissioning vis a vis man-rem budgeting.

Dismantlement methods and the procedures employed now will form the basis for the selection of these for decommissioning activities as well as lead to improvements of the existing techniques from a cost, time and ALARA viewpoint.

Data will be stored in a database system for decommissioning. The normal practice of internal reports will add to this data. Also, the existing system of record keeping in hard copies in fireproof cabinets in a records room ensure availability of the records at any future date.

9.2. Pending issues

The refurbishment experience and the CRP have identified the need to further work on the cleanup of contaminated soil, improved methods for radioactive sump cleaning, large scale decontamination facilities, automated cutting and dismantling methods, on line characterization instruments, sufficient storage facilities for radioactive wastes, suitable volume reduction methods for decommissioning wastes that cannot be decontaminated to exempt levels, techno-economic studies of decommissioning activities and budget provisions. Work on some of these activities has already begun.

9.3. Decommissioning activities

The CIRUS reactor is designed to facilitate the replacement of the reactor vessel. This feature makes the removal of components above the reactor vessel simpler without remote cutting etc. However it is to be expected that the dowel pins, which locate the shields, one above the other, may be jammed in their holes. This is based on similar experience at NRX, Canada, where these steel thermal shields got jammed. Removal of these shields could pose some difficulties, which may be overcome by proper planning.

A conceptual decommissioning programme for CIRUS will include:

- Planning based on the above data, which will categorize systems, structures and components into ones that can be decontaminated to exempt levels and the remaining ones, which have to be handled as radioactive waste.
- Final shutdown and defuelling.
- Shipping fuel out of the reactor complex.
- Recirculating the process fluids through on-line ion-exchange streams to reduce the radionuclides in the systems to as low as possible before draining the systems.
- Dismantle, decontaminate and release the components that can be decontaminated to exempt levels.
- Dismantle out-of-core components and release them for waste conditioning and disposal.
- Defer dismantling of in core components until radiation levels are reduced to affordable levels.
- Dismantle the reactor structure sequentially by removing the master plate, biological shields, upper thermal shields and reactor vessel separately, each as “one piece” with the remotely operated crane and transfer them for disposal.

- Remove remaining components from the reactor structure except for the radial biological shields.
- Remove the graphite reflectors and send for conditioning viz. annealing and disposal.
- Characterize the concrete structures and decide on strategy i.e. either segregate active and inactive or dispose as active.
- Clean up the spent fuel storage bay after sufficient decay of radioactive debris.
- Clean sumps, collect core samples, break lining and demolish till activity levels are to that of exempt levels.
- Detailed surveys of soil around the reactor complex, segregate active and inactive soil and dispose of the active soil as radioactive waste.
- Rebuild and relandscape.
- Methods for the final radiation surveying of the site and release for either restricted or unrestricted use.

10. Conclusion

CIRUS has been operated with good fuel performance. Consequently radiation levels on out-of-core systems, structures and components; such as heat exchangers, primary coolant pipelines, concrete water storage tanks, etc. are very low, as expected. These will not pose significant problems during decommissioning. In-core components show high radiation levels due to activation products mainly ^{60}Co . An appropriate plan of action for dismantling and disposal of these components/structures will be needed.

The experience and expertise gained and lessons learned during the on-going refurbishment work, will be a useful input for working out appropriate decommissioning strategies for CIRUS at a later stage.

In addition, it is expected, that the data that has been generated now: the radiation surveys of the reactor structure; radiological characterization of various components, systems and structures; experience in dismantlement of piping and heat exchangers; decontamination trials; disposal of dismantled equipment and piping; clean up operation of various facilities; development of remotely operated equipments and experience with remote repairs etc. will aid in preparation of detailed plans with optimum man-rem budgeting and will give operational guidance to facilitate future decommissioning of CIRUS in a safe manner.

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Decontamination of TRIGA Mark II reactor, Indonesia

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Abstract. The TRIGA Mark II Reactor in the Centre for Research and Development Nuclear Technique Bandung has been partially decommissioned as part of an upgrading project. The upgrading project was carried out from 1995 to 2000 and is being commissioned in 2001. The decommissioning portion of the project included disassembly of some components of the reactor core, producing contaminated material. This contaminated material (grid plate, reflector, thermal column, heat exchanger and pipe) will be sent to the Decontamination Facility at the Radioactive Waste Management Development Centre.

1. Introduction

This contribution to the IAEA Co-ordinated Research Project (CRP) on “Decommissioning Techniques for Research Reactors” presents a slightly different aspect of decommissioning than is normally considered. The paper describes the partial decommissioning of the present Bandung TRIGA Mark II reactor, to make way for upgrading the facility. The extent of the decommissioning in this case, and in many other upgrade projects, requires essentially the same degree of effort and preparations, as would a complete decommissioning without the upgrade. For this reason the preparations and planning for this project parallel that needed for a final decommissioning project that involves a deferral stage. The upgrade effort can be considered as two separate efforts; the decommissioning and the rebuilding. Both require appropriate licencing actions, plans, schedules, and detailed information

The National Nuclear Energy Agency (BATAN) possesses three research reactors, located in three different cities: Bandung, Yogyakarta and Jakarta (Serpong). Normally the Bandung reactor is utilized for the production of radioisotopes and radiopharmaceuticals as well as for carrying out research in neutron radiography, neutron activation analysis and nuclear instrumentation. In cooperation with Bandung Institute of Technology education and training programmes are conducted for the purpose of man power development. The Kartini reactor in Yogyakarta is used primarily for conducting education and training programmes in cooperation with the Gadjah Mada University. Research and development in radioactive waste management and refining of nuclear materials and nuclear instrumentation has been carried out in the Yogyakarta Nuclear Research Centre. The Multipurpose G.A. Siwabessy reactor (in Indonesia called RSG-GAS), located at Serpong, is a high flux research reactor with average thermal neutron flux of $2.5 \times 10^{14} \text{ n/cm}^2/\text{s}$, fueled with $\text{U}_3\text{O}_8\text{-Al}$ with LEU (low enrichment uranium — less than 20 % ^{235}U in U). The reactor is cooled and moderated by light water, with a reflector of Beryllium assemblies. The RSG-GAS is a multipurpose reactor with a high neutron flux and a number of irradiation facilities to optimize its utilization.

The Bandung reactor type is a TRIGA Mark II type that began operation in 1965 at a power level of 250 kW. In 1971 the reactor power was upgraded to 1000 kW, and operated at that power level between 1971 and 1996. The main modifications in this upgrade were the core structure and the reactor cooling system. In 1994 the instrumentation and control system

was replaced with a digital system. This system enables the reactor to be started up automatically, provides more accurate and complete operational data, and enhances the safety of reactor operation. In early 1995 the National Nuclear Energy Agency of Indonesia (BATAN) decided to upgrade the reactor power level from 1000 kW to 2000 kW. The reasons for the upgrade include [1]:

- (1) to increase the operational safety of the reactor. The increasing of operational safety can be achieved by modifications to the integrated cooling system (the primary cooling subsystem, diffuser subsystem and water purification subsystem) to create separate subsystems. The purpose of modification is to simplify the operation, control and maintenance of these systems.
- (2) to address the increase in demand for radioisotopes, and enhance the reactor's ability to continue to supply BATAN, especially for Mo-99.

2. Decommissioning planning and implementation of the upgrade

Based on act number 10/1997 on "Nuclear Energy", the regulatory body related to nuclear energy became BAPETEN [2]. On the other hand, BATAN remained as an executor of research, development and application of nuclear energy. Consequently, permission to decommission a research reactor must be approved by BAPETEN. According to the act, the owner of the nuclear installation, as well as the research reactor operators must have permission for construction, operation and decommissioning of a research reactor, based on this regulation BATAN is the owner of the facility. Therefore, the planning and implementation of the partial decommissioning of reactor the TRIGA Mark II for upgrading purposes was done by the Centre for Research and Development Nuclear Techniques Bandung. In anticipation of the decommissioning programme for the research reactor as well as other nuclear facilities, BATAN established the Radioactive Waste Management Development Centre at the Serpong site, near Jakarta. One of the purposes of this centre is to do research and development on decontamination and decommissioning of nuclear devices or nuclear facilities. The upgrading project started in February 1996 and was completed in March.

The decommissioning planning was done, in anticipation of the upgrade, according to the Introduction Report of Safety Analysis for Upgrading of Bandung TRIGA Mark II Reactor. The decommissioning plan includes the following sections;

Introduction

Description of Facility; Radioactive Inventory

Selection of Decommissioning Technology

Organization; Time Scheduling

Cost and Budgeting

Decommissioning Activity

3. Description of facility

The extent of the upgrading includes: modification of the reactor tank, reactor core, reflector, primary cooling system, secondary cooling system, control system, etc. Previously, the reactor tank was 6553 mm high, 1981 mm in diameter, and 6 mm thick; made from the aluminium alloy 60-61-T651. The tank will be changed to a height of 7553 mm, with the

same diameter, thickness and material type. The core configuration will be changed from an annular form to a hexagonal form, and the number of control rods will be increased from 4 to 5. Another modification will be the fuel. Previously, there were two types of fuel used, containing uranium enriched to 8,5 % and 12 %. The new fuel will contain uranium enriched to 20%. Calculations indicate that after operating for 1530 full power days, the core will contain 100 fuel elements of the 20% enriched uranium type. This core size will then be maintained for the lifetime of the reactor.

The new core will contain five irradiation positions. One in the centre thimble, one for a pneumatic transfer system, and three for the irradiation of targets for Mo-99 production. The centre thimble can be changed by irradiation target. The neutron thermal flux for the five positions was calculated to be between 1.1×10^{13} and $3,8 \times 10^{13}$ n/cm²/s. The graphite reflector will be replaced and the old one segregated into solid active waste. Table I presents data on the replacement programme for upgrading of the TRIGA Mark II.

4. Radioactive inventory

The radioactive inventory consists of three types of items:

- Fuel elements;
- Reactor core components; and
- Auxiliary systems components.

Data for the fuel elements of the TRIGA Mark II Reactor are shown in Table II. The fuel elements will be reused after the upgrading project is completed. Any spent fuel elements will be returned to the supplier.

Table III shows data from radiation level measurement made in the Reactor Core zone after the fuel elements were removed.

The reactor core components consist of: Top grid, Bottom grid, Lazy Susan and Bellows. The measured radiation fields from each component are between 10 and 40 R/hr. The radiation fields measured at the Lazy Susan are given in Table IV.

Table V presents radiation level and activity data for some of the auxiliary system components: primary cooling water heat exchanger (PHE); secondary cooling water heat exchanger (SHE); and pipes.

5. Decommissioning activity

5.1. Dismantling

The removal of the core began with the removal of the fuel elements from the reactor core to the bulk-shielding tank and to the spent fuel storage pit. Then the work continued with a visual inspection of the thermal column and maintenance of the bulk-shielding tank. Afterward, the rotary specimen rack, neutron detector and USIF were also removed from reactor tank and stored in the bulk shielding tank. Following this the top and bottom grid plates as well as the safety plates were disassembled and stored in the bulk-shielding tank. Next the bellows was removed from the reflector. Finally the graphite reflector was removed from the bottom the reactor tank.

Table I. Design changes for upgrading the TRIGA Mark II reactor

System and Components	Parameter	Previous Design	Upgrade Design
Reactor			
Concrete	Outside diameter	6969 mm,	The same
	Inside diameter	2438 mm	The same
	Height	8636 mm	The same
Reactor tank	Height	6553 mm	7553 mm
	Diameter	1981 mm	<1981 mm**
	Thickness	6 mm	same thickness
	Volume	30 m ³	35 m ³
Reflector	material	graphite	replaced — same
Thermal Column and Beam ports (4)	aluminium casing material	Al 5052	Al 60-61-T651.
	thickness	1 inch	same thickness
Core	diameter	560 mm	same diameter
	height	1000 mm	same height
	# of core positions	127	121
	# of fuel elements	110	100 (after 1530d)
	configuration form	annular	hexagonal
	uranium content	55 g/fuel element	98 g/ fuel element
	enrichment	12 % and 8.5%	20%
	grid plate		matched
Bellows	dimensions	500mm, 250 mm	replaced — same
Primary Cooling System			
Piping	material	SS-316	replaced — same
Heat exchanger	capacity	360 gpm	720 gpm
Water filter	filtration	10 μ and 50 μ	replaced — same
Ion exchanger	volume	300 l	600 l
	capacity	360 gpm	720 gpm
Secondary Cooling System			
Piping	material	carbon steel	replaced — same
Cooling tower	capacity	720 gpm	1440 gpm
VAC/ Off-gas System			
HEPA filters	dimensions	60×220×160 mm	replaced — same
	number	144	144

** The new tank was inserted in the old tank and the space between was filled with concrete.

Table II. Fuel element data for TRIGA Mark II

Item	Type	Number
Fuel elements	Type 104	97
	Type 106	95
Instrumented Fuel elements (IFE)	Type 204	1
	Type 206	3
Fuel Follower Control Rods	Type 304	4
	Type 306	4
Total		204

Table III. Radiation measurements in core zone, FUEL removed

No	Location	Radiation Level
1	Surface of Bellows	130R/hr
2	Surface of Bellows	100R/hr
3	Surface of Bellows	120R/hr
4	Surface of Clem Bellows	150R/hr
5	Surface of Reflector Pipe	110R/hr
6	Surface of Clem Bellows	110R/hr
7	Surface of CT	560R/hr
8	Surface of Reflector	66R/hr
9	Surface of Reflector Pipe	70R/hr
10	Surface of Reflector Pipe	70R/hr
11	1 m from Bellows	90R/hr
12	Surface of Fuel Rack	8R/hr
13	Grid (CT parallel)	450R/hr
14	1 m (under grid)	80R/hr

Table IV. Radiation levels at Lazy Susan

No	Location	Radiation Level
1	Surface of LS(North-up)	1000 R/hr
2	Surface of LS(West-up)	1100 R/hr
3	Surface of LS(East-up)	1200 R/hr
4	Surface of LS(South-up)	1000 R/hr
5	Surface of LS(West-in)	1800 R/hr
6	Surface of LS(North-in)	2400 R/hr
7	Surface of LS(East-in)	2100 R/hr
8	Surface of LS(South-in)	2400 R/hr
9	Surface of LS(West-in, rotary)	2500 R/hr
10	Surface of LS(hole)	2200 R/hr

Table V. Radiation levels and activities of PHE, SHE & pipes

No	Item	Radiation Levels ($\mu\text{Sv/hr}$)	Activities (Bq/cm^2)
1	Surface of PHE (front)	0.30	6.1
2	Surface of PHE (top)	0.40	7.1
3	Surface of PHE (bottom)	0.40	18.5
4	Surface of PHE (behind)	0.40	2.1
5	Surface of PHE (side)	0.40	18.5
6	Surface of PHE (front cover)	0.25	2.1
7	Surface of PHE (behind cover)	0.25	16.4
8	Surface of SHE (front)	0.25	none
9	Surface of SHE (side)	0.25	none
10	Surface of SHE (behind)	0.20	none
11	Pipes	0.25–0.30	2.1–8.2

According to the report from the Centre for Research and Development Nuclear Techniques Bandung [1], the program for upgrading the reactor included the following activities: inspections, removal and disassembly of components. The disassembly activities are listed below:

- (1) Removal of the fuel elements from the reactor core to the bulk shielding tank and the spent fuel storage pit.
- (2) Disassembly of the cooling system (primary cooling subsystem, heat exchanger subsystem, diffuser subsystem and water purification subsystem).
- (3) Disassembly of the reactor core components (control rods, grid plate, rotary specimen rack, pneumatic transfer system, bellows, detector support, reflector).
- (4) Removal of the graphite from the thermal and thermalizing columns.
- (5) Disassembly of the thermal and thermalizing columns.

The main problems encountered during these operations are summarized in Table VI.

5.2. Decontamination

According to the inventory data, the radioactive wastes consist of high level radioactive wastes, low level radioactive wastes, including extremely low level radioactive wastes and non-radioactive wastes. The high level radioactive wastes were managed by placing them in the temporary interim storage. In certain cases, solid high level radioactive waste was decontaminated to lower its activity for easier handling and storage in the available facility. The low level radioactive wastes and extremely low level radioactive wastes were also placed in the interim storage. All solid waste that originated from the reactor core is listed in Table VII.

The contamination of core components may consist of activated corrosion products, fuel fragments and/or fission products. Normally, the radionuclide of most concern is ^{60}Co , primarily as a result of the activation of stainless steel components. While the aluminium reactor core components contain little long-lived activity, the stainless steel parts such as bolts and nuts, etc., may have a high radiation field associated with them. Other radionuclides of concern are ^{54}Mn , ^{65}Zn and ^{55}Fe . Also, Tritium and ^{14}C may be found in graphite, due to the

presence of lithium and nitrogen impurities [3–5]. All core components will be decontaminated by a harsh chemical solution to remove surface contamination and some of the component activation products at the component surface. Since all components have been removed and stored, their future disposition planning will consider decontamination factors and the quantity of secondary waste generated.

Table VI. Main problems encountered and solutions found

System	Operation	Problem	Solution
Tank and reactor core	Removal of the fuel elements from the reactor core to the bulk shielding tank and the spent fuel storage pit.	None	None
	Disassembly of the Lazy Susan, bellows, pneumatic transfer system and reflector.	High exposure to personal	Pool water and lead provided as shielding
	Removal of graphite from the thermal and thermalizing columns.	Space of thermal and thermalizing column too small	Drilling out some graphite blocks and pulling out by tracker.
	Disassembly thermal and thermalizing column.	Height exposure to Personal	Dismantling column by remote plasma cutting.
	Inspection of the tank	Heavy corrosion on tank so it cannot be reused.	Insert new tank into the previous tank
Primary Cooling System	Disassembly of cooling system (primary cooling subsystem, heat exchanger subsystem, diffuser subsystem and water purification subsystem).	None	None
Secondary Cooling System	Disassembly of cooling tower, secondary pumps, piping and heat exchanger	Mobile crane not available	Rented one from another site.

Table VII. Core replacement components

Item	Radiation Levels (R/hr)	Material	Treatment before decision for decontamination
Grid plate	1	Aluminum	Storage
Reflector	28	Aluminium + graphite	Storage in water bath
Thermal and Thermalizing column	3.5	Aluminium	Dismantle into smaller parts

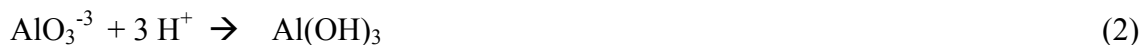
Components containing extremely low level radioactive wastes, such as the PHE, could be decontaminated to meet non-radioactive waste criteria, so they become available for unrestricted reuse. Normally, the radionuclide of most concern is ^{60}Co , primarily as a result of the activation of stainless steel components. Cesium-137, ^{152}Eu and ^{154}Eu are fission products and may be found on the internal surfaces of the primary circuit [3]. Aggressive methods are intended to remove contamination along with a substrate layer.

The plan calls for the components to be decontaminated at the decontamination facility before being dismantled. Laboratory experiments must be carried out to investigate and optimize the chemical decontamination process for this purpose. The experiments will work with aluminium and stainless steel materials representative of the core components, heat exchanger and pipes.

5.2.1. *Simulation chemical decontamination process for reflector*

The reflector is made from graphite encased in a can made from the Aluminium alloy, Al 60-61-T 651. The radiation field from the reflector, 28 rad/hr, arises from activation and contamination of the Aluminium alloy and stainless steel bolts in the reflector. The dominant activation nuclides from Aluminium and stainless steel are ^{60}Co , ^{54}Zn , ^{65}Zn , ^{55}Fe and ^{63}Ni . The dominant nuclides from fission product contamination left on the reflector surface is ^{137}Cs . The nuclides formed during the activation of aluminium are ^{54}Mn , ^{65}Zn and ^{55}Fe , which are not of concern following long decay periods because they have short half-lives (0.9 years for ^{54}Mn , 0.7 years for ^{65}Zn and 2.7 years for ^{55}Fe). The ^3H and ^{14}C activity in the graphite does not contribute to the surface dose.

Although the reflector is not being reused, it will be decontaminated in order to decrease the doses rate and thus reduce the radiation exposure of the labourers, making it easier to store in the available facility. A chemical decontamination method by NaOH solutions and/or oxalic acid will be investigated for use for this task. The reaction on the oxide surface layer is the following:



During this process will be oxidized by the strong reaction:



For the reaction using oxalic acid the basic metal will be ionized as follows:



Technical approach

The Al 60-61-T651 specimens used in the investigation were 50mm × 50mm × 1mm (the materials and thickness of the specimens was the same as for the reflector covering material). The decontaminating agent used was NaOH in various concentrations (1, 1.5, 2 and 3%). This was followed by a rinse containing HNO₃ (1 and 2%) or H₂C₂O₄ (2 and 5%). The times for the decontamination step were 30, 60, 90, 120, 150 and 180 minutes. The corrosion rate was calculated by the difference in the specimen weight before and after decontamination.

This was corroborated by Atomic Absorption Spectrophotometry data of the dissolved Al in the NaOH solutions.

Accomplishment

The results of the investigation are summarized in Table VIII and Table IX. Table VIII presents the visual appearance of the surface material after the chemical decontamination process. Table IX gives the corrosion rate as a function of the concentration of NaOH and HNO₃ for a given decontamination period of 180 minutes. According to the task effectiveness data, the optimum concentration is about 3% NaOH and 2% HNO₃ and the operation time 180 minutes.

Table VIII demonstrates that there is no significant difference between rinsing with HNO₃ or with H₂C₂O₄. The parameters chosen for the actual decontamination of the reflector are: a 3% NaOH solution; a 2% HNO₃ solution, room temperature and a time of 180 minutes.

5.2.2. Simulation chemical decontamination process for PHE

The radioactivity in the primary cooling water heat exchanger (PHE) is about 18.45 Bq/cm² (Table V). Since the limit for it to be considered as non radioactive is 4 Bq/cm², the PHE must be decontaminated before it can be released for reuse or sent to a landfill for disposal.

Table VIII. Appearance of Al 60-61-T651 after decontamination for 90 minutes

No	NaOH, %	HNO ₃ , %	H ₂ C ₂ O ₄ , %	Visual Appearance
1.	1	1	-	jet black
2.	1½	1	-	jet black
3.	2	2	-	shiny
4.	3	2	-	shiny
5.	1	-	2	jet black
6.	1½	-	2	jet black
7.	2	-	5	shiny
8.	3	-	5	shiny

Table IX. Corrosion rate of Al 60-61-T651 decontaminated with NaOH-HNO₃

No	NaOH, %	HNO ₃ , %	Decon. time (minutes)	Corrosion rate (µm)
1	1	1	180	56.10
2	1½	1	180	58.26
3	2	2	180	85.30
4	3	2	180	159.83

The chemical decontamination method investigated for this purpose was a combination of HNO₃, and KMnO₄, rinsed with H₂C₂O₄ as well as KOH, KMnO₄ and H₂C₂O₄. The reaction employed is:



The first part is the transfer of an electron from the reducing agent to the oxide's metal ion to produce an unstable reduced species, and the second one is the removal of this reduced ion and associated oxide anions into solution.

Another reaction is



Before starting the actual decontamination of the PHE experiments were carried out with simulated samples.

Technical approach

The specimens used in the investigation were 50 x 50 x 1 mm pieces of stainless steel 316, contaminated with about 30 Bq/cm² of ¹³⁷Cs. The decontaminating agents used were HNO₃ (1, 2, 3 and 5%) and KMnO₄ (0.1, 0.2 and 0.3%) combined with 2% H₂C₂O₄. The decontamination factor was measured as function of concentration of HNO₃ and KMnO₄. A NaI-Tl detector coupled to a MCA counted the initial and final activity of the specimens.

Accomplishment

The results of the investigation are summarized in Table XI and Table XII. Table XI shows the influence of the concentration of HNO₃ and KMnO₄. Table XII shows the influence of concentration of KOH and KMnO₄. It can be seen from the tables, that the application of an acidic solution of potassium permanganate is more effective than the application of a basic solution of the same oxidation agent. The decontamination of the PHE will use a solution of 5% HNO₃, 0.3% KMnO₄ and 2% H₂C₂O₄, and at a temperature of 60°C.

TABLE X. Influence of concentration of HNO₃ and KMnO₄*

No	Concentration HNO ₃ , %	Concentration KMnO ₄ %	DF
1	1	0.1	60
2	2	0.1	130
3	3	0.1	230
4	5	0.1	310
5	1	0.2	100
6	2	0.2	200
7	3	0.2	260
8	5	0.2	400
9	1	0.3	100
10	2	0.3	250
11	3	0.3	300
12	5	0.5	500

* Temperature: 60°C; Time: 1 hr; Concentration of H₂C₂O₄: 2%

TABLE XI. Influence of concentration of KOH and KMnO_4 *

No	Concentration KOH, %	Concentration KMnO_4 , %	DF
1	1	0.1	10
2	2	0.1	20
3	3	0.1	25
4	5	0.1	40
5	1	0.2	20
6	2	0.2	20
7	3	0.2	35
8	5	0.2	40
9	1	0.3	20
10	2	0.3	35
11	3	0.3	40
12	5	0.3	60

* Temperature: 60°C; Time: 1 hr; Concentration of $\text{H}_2\text{C}_2\text{O}_4$: 2%

6. Planning for the decontamination of dismantled components at the radioactive waste management development centre

Selected dismantled components will not be decontaminated at the Bandung site but will be sent to the decontamination facility at the Radioactive Waste Management Development Centre. At the Radioactive Waste Management Development Centre (RWMD) there are several methods available for radioactive waste management as well as for decontamination of nuclear devices such as; sand blasting, chemical immersion, ultrasonic cleaner, water jet, chemical treatment for liquid waste, incinerator, and interim storage for low and intermediate level waste. All dismantled components will be sent to RWMD by solid waste transporter. Considering that a solid waste transporter can only be used for low- and intermediate-level waste then the grid plate and thermal and thermalizing column will be sent early to RWMD. The reflector can not be sent immediately since it is high-level waste. The reflector will remain in storage at the Bandung site until the radiation level has decreased enough to classify it as intermediate-level waste. The dismantled piping and the thermalizing column will be decontaminated in a 1000L chemical immersion tank that measures $200 \times 60 \times 110$ cm. To accelerate the decontamination process the immersion tank is connected via a piping system to a boiler and circulation pump that can recirculate the decontaminating solution at a flow rate of 750 L/hr. Small pieces of dismantled components will be decontaminated in an ultrasonic cleaner unit that contains 2 generators, 2 transducers of 40 kHz and one $51 \times 46 \times 50$ cm immersion tank. The heat exchanger will be decontaminated by closed-loop circulation of the chemical solution.

All secondary liquid waste will be treated by chemical precipitation in a chemical waste treatment unit. After removing the precipitation the liquid will be treated in an evaporator which has an operating capacity of $0.75\text{m}^3/\text{h}$.

7. Conclusions

- The up-grading of TRIGA Mark II Reactor gave specific lessons learned.
- Some radioactive waste from the reactor removal can be released for unrestricted use or disposed of as landfill material, other types of radioactive waste must be managed, and still others must be decontaminated before being reused or disposed of as landfill material.
- Following decontamination to simplify storage, the reflector will be stored at the Serpong site. A decontamination solution of 3% NaOH and 2% HNO₃, will be applied at room temperature for 180 minutes.
- The primary cooling water heat exchanger (PHE) must be decontaminated before reuse. A decontamination solution of 5% HNO₃, 0.3% KMnO₄ and H₂C₂O₄ will be used.

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Studies on decommissioning of TRIGA reactors and site restoration technologies in the Republic of Korea

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Abstract. Research and development on research reactor decommissioning and environmental restoration has been carried out at KAERI since 1997 to prepare for the decommissioning of KAERI's two TRIGA-type research reactors, which had been shut down since 1995. A 3-D graphic model of the TRIGA research reactor was built using IGRIP. The dismantling process was simulated in the graphic environment to verify the feasibility of individual operations before the execution of the remote dismantling process. An under-water wall-climbing robot, moving by propeller injection, and identifying its coordinates by using a laser sensor, was developed and tested in the TRIGA reactor pool by measuring a radioactive contamination map of the reactor surface. Using MODFLOW and TRIGA site geological data, a computer simulation of the underground migration of residual radionuclides, after the TRIGA reactor decommissioning, was carried out. It was found that the underground migration rate was very slow such that, when radionuclide decay and dilution are considered, the residual radionuclides will not have a significant environmental impact. The soil decontamination R&D, using soil washing, solvent flushing and electro-decontamination technologies, was carried out to determine the best method for decontaminating the soil waste accumulated in KAERI. The decontamination results indicated that, using the soil washing method, more than 80% of the soil wastes could be decontaminated well enough to discharge them to the environment. It was also determined that the control of solution pH and temperature in the soil washing process is important for the reduction of decontamination waste. Further decontamination, using an electro-kinetic decontamination method, was considered necessary for the residual soil waste, which consisted mainly of fine soil particles.

1. Objective

The first research reactor in Korea was a TRIGA Mark-II type, which began operation in 1962. The second one, a TRIGA Mark-III model, began operation in 1972. Both reactors had their operations phased out in 1995 due to their age and the commencement of operation of the new research reactor, HANARO, at the Korea Atomic Energy Research Institute (KAERI) in Taejeon. The decommissioning project of the two TRIGAs was begun in January, 1997 and will be completed in December, 2007. The goal of this project is to decommission the reactor site completely enough to allow its release for unrestricted reuse and to reduce the volume of the resulting decommissioning wastes, including soil, to an as low as reasonably achievable level.

The objective of the R&D program was to develop the decommissioning and environmental restoration technologies necessary, not only for the TRIGA reactor decommissioning project, but also for the future needs in the country, by using the retired TRIGA reactors as experimental objects for testing and technology demonstration.

2. Computer simulation of research reactor dismantling process

Remote dismantling of nuclear facilities is desirable to shorten working time in the radiation environment and to reduce human exposure. Before the execution of remote dismantling processes; however, it is essential to verify the feasibility of the individual operations through realistic graphic computer simulation. In this work, therefore, a 3-D graphic model of a research reactor is built and its dismantling process is simulated in the graphic environment.

2.1. *Design of dismantling process*

In reactor dismantling, the components to be cut or disassembled include the reactor tank, reactor internals, components and support structures around the reactor, piping, tank, machine parts and concrete walls. The types of equipment that can be used for the various dismantling processes are summarized in Table I.

Table I. Dismantling processes and equipment

Dismantling Process	Tools	Manipulator
Removal of Reactor peripherals	Plasma arc cutter	Crane
Removal of Reactor Internals	Rotary disk knife Shaped explosives	Manipulator
Removal of Reactor Core	Plasma arc cutter	Crane Manipulator
Removal of Reactor Structures	Manipulator Shear cutter	Crane Manipulator
Removal of Reactor Tank Liner	Arc saw	Crane
Demolishing of Concrete walls	Water jet Diamond saw Controlled blasting	Crane Excavator

2.2. *Graphic simulation system*

As a graphics tool, the project used a commercial software package, IGRIP (Interactive Graphics Robot Instruction Program), supplied by Deneb Robotics Company. This programme requires the use of a high performance computer that is capable of performing real-time graphic animation and computation. The graphic simulation system is composed of a facilities and equipment modelling program, a simulation program and a program allowing connections to external equipment. Figure 1 illustrates the structure of the graphic simulation system.

2.3. *Graphic modelling of research reactors*

Research reactor facilities and dismantling equipment were drawn in 3-D CAD models. Assembly size and shape of all models coincide with actual design drawings, and standard coordinates are assigned for easy assembly. Each assembly part is modelled with independent mobility. Each part is also assigned various mobility attributes such as the relative assembly position, the kinematic constraints, and range of mobility. In concordance with such purpose, IGRIP is provided with the function of reverse kinematics, dynamic, and liaison conditions between the parts. As a result, the graphic models are constructed as independently operating entities.

The graphic model of the reactor is composed of its various parts: reactor peripherals, centre channel, reflector and specimen rack, fuel elements, reactor tank, internal tubes, thermal column and thermal column door, etc., as illustrate Figure 2.

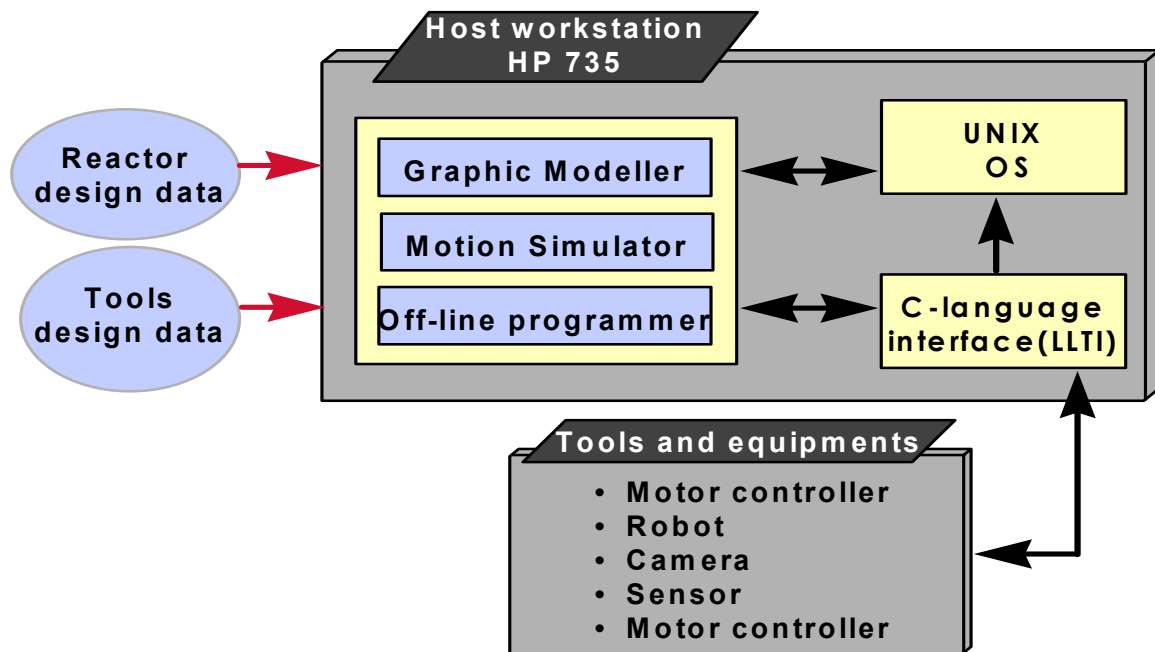
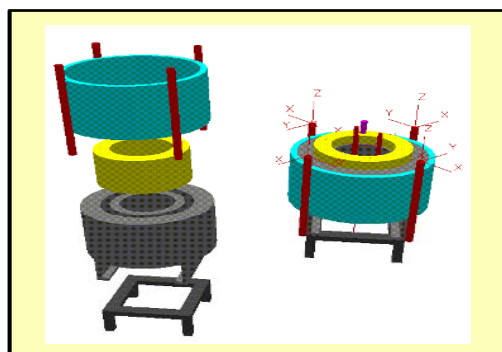
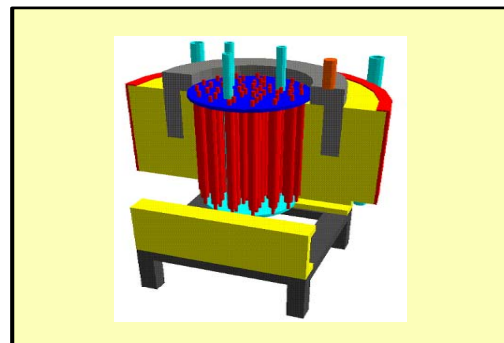


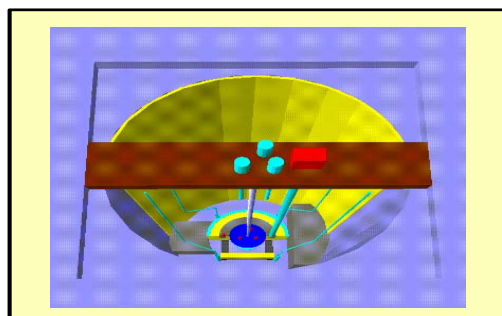
Figure 1. Configuration of IGRIP graphic simulation system.



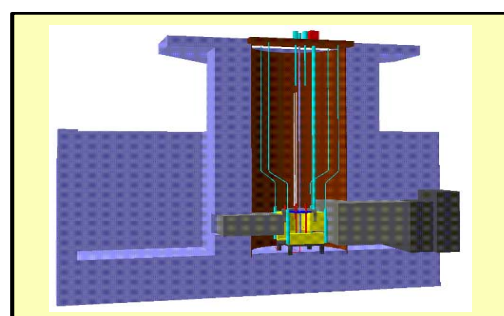
(a) Core and reflector



(b) Fuel element



(c) Center channel



(d) Reactor internals

Figure 2. Graphic model of TRIGA research reactor (Republic of Korea) by IGRIP.

Each structural part is drawn in cross-section — the right and left section. During the simulation, therefore, making one of the half-sectional parts transparent can make the internal view of the facility visible. The models of dismantling equipment include the crane, bridge transporter and robot. This equipment is mounted on the research reactor model in proper arrangements to compose the entire simulation environment.

2.4. *Computer simulation of dismantling process*

Using the graphic models, a dismantling process is assimilated via graphical animation. Such a simulation can effectively be used to study the dismantling process without actually performing it, thus reducing the time and effort required for process design. Simulation is performed on the removal of environmental devices, internal piping, thermal column, reactor tank liner and the dismantling of installed equipment, etc. With the simulation, each dismantling process is optimized through iteratively checking the validity of various locations of the robot and its movement paths, and verifying that there are no interferences with other items. The robot's movement paths are created as a sequential series of tag points, which are tracked by the robot's end effector.

The simulation tool is programmed with Graphic Simulation Language (GSL) provided as a part of IGRIP. GSL can simultaneously direct all equipment operation in the working area, while continuously auditing the interference of each component during the operation. The process of computer simulation thus incorporates a graphic display and kinematic and dynamic calculations; all performed at the same time, demonstrating the progress of the work process in real time. The typical graphic simulation results for the TRIGA reactor dismantling processes are shown in the Figure 3.

2.5. *Conclusion*

The 3-D graphic simulation of a dismantling process, as presented in this work, can be used as a supplementary technology for dismantling of nuclear facilities as a means to verify the dismantling process. This technology may also function as an effective aid for remote dismantling processes, not only in process design, but also in on-site execution of remote dismantling processes.

3. Development of remote wall climbing robot for inspection and decontamination

3.1. *Wall climbing robot function*

The dismantling of the research reactors required an underwater measurement of the surface contamination level of the reactor tanks, 2.0m(D) × 6.2m(H) in case of TRIGA MARK III, and decontamination of any hot spots, for example in the bottom of the reactor. The underwater wall-climbing robot was developed to measure the surface contamination levels on components in the water and thereby to obtain the contamination map remotely and automatically. Figure 4 shows the conceptual drawing of the wall-climbing robot carrying out the inspection and decontamination of the reactor pool.

3.2. *Selection of the wall climbing robots technologies*

Wall adhesion methods using magnetic or vacuum technologies, and autonomous navigation ability are prerequisites to the development of an effective wall-climbing robot. Typical models available and their specifications are summarized in Table II. With reference to the table, wall-climbing robots were developed for applications in many industrial facilities such as buildings, warehouses, etc.

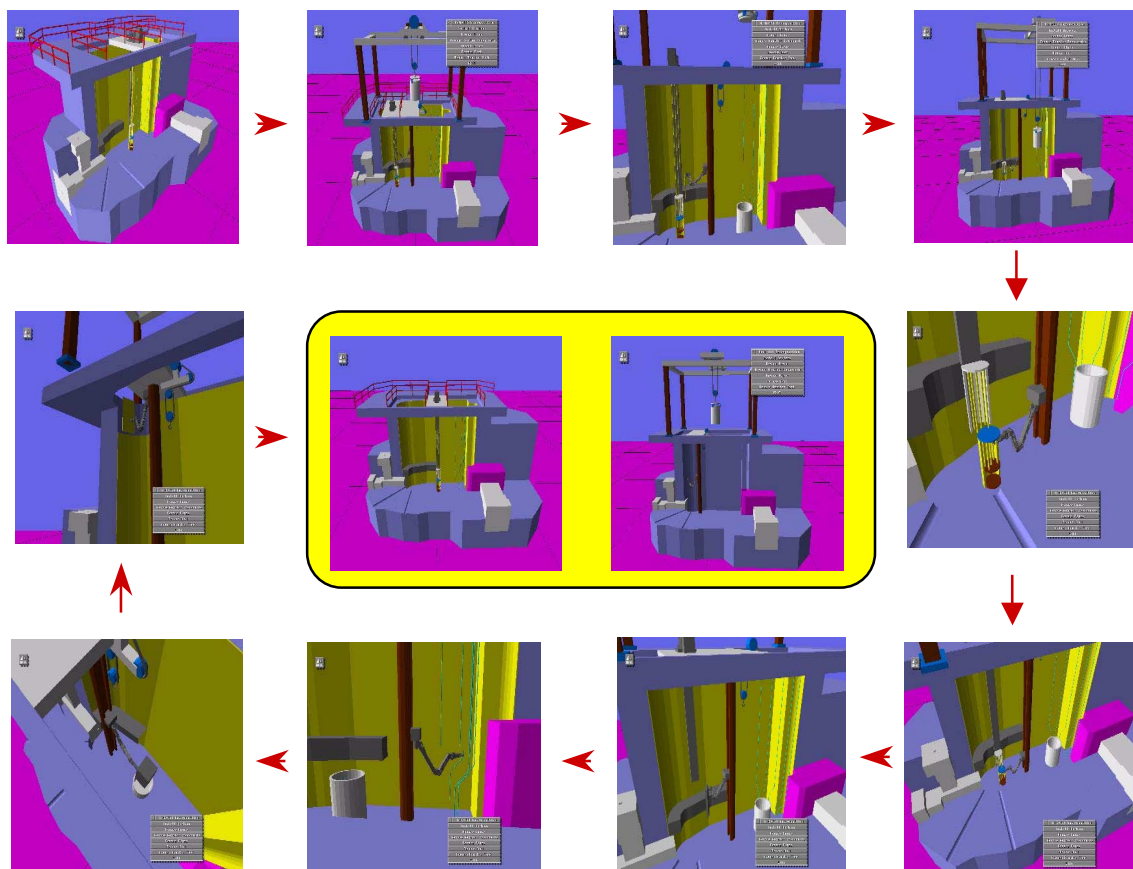


Figure 3. 3 dimensional graphic simulator for investigation of the feasibility of decommissioning process.

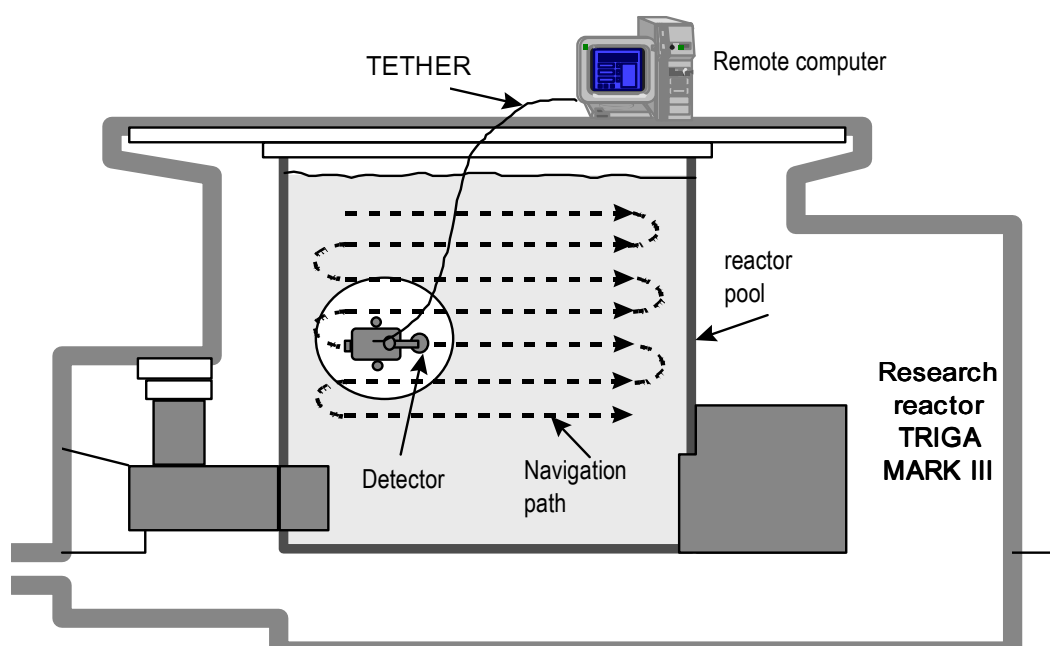


Figure 4. Inspection and decontamination using wall-climbing robot.

However, these robots are unable to be directly applied to the underwater inspection and decontamination of research reactors because they were designed to be operated in air.

3.3. *Development of the underwater wall climbing robot design concept*

The working environment and the design specifications of the wall-climbing robot for use in the TRIGA reactor are:

- (1) the working environment;
 - working environment: under water (robot sealing required)
 - wall material: concrete and metal (magnet wheel adhesion precluded)
 - wall structure: rectangular (ranging function on the wall edge required)
- (2) the design specifications;
 - ranging with constant velocity in x and y direction
 - maintaining a certain distance between wall and robot
 - compensating robot weight by buoyancy (approximate coincidence of centre of weight and centre of buoyancy)
 - maximum ranging velocity > 300 mm/sec (maximum velocity of the existing wall climbing robots is 170 mm/sec)
 - size: less than 400(W) x 400(L) x 500(H) (considering the operation on edges)
 - equipment and sensors installed: camera, lighting, gyroscope, integration controller contamination measurement system, decontamination tool, and two axes manipulator

Table II. Wall-climbing robots and their characteristics

Model	Characteristic	Specification
NINJA-I by S. Hirose, Tokyo Inst. of Technology (1991)	Legged Type 4 Legged+ Vacuum Suction Legs: 3D parallel, Motor Actuated Vacuum Suction	Weight : 45 kg Size : 1.8 x 0.5 m Maximum Speed : 16 cm/sec
ROBUG-II by Collie, Portsmouth Polytechnic (1991)	Legged Type 4Legged+ Magnet, Vacuum Suction Legs: 3D Parallel, Cylinder	Weight : 17 kg Size : 1.0 x 0.7 m Maximum Speed : N/A
Biped Walking Robot by A. Nishi, Niyazaki Univ. (1992)	Legged Type 2 Legged+ Large Vacuum Suction Motor Actuated	Weight : 45 kg Size : 1.8 x 0.5 m Maximum Speed : N/A
CEIT(prototype) (1994)	Worm Type 3 Vacuum Sucker Attached Cylinder Actuated	Weight : 45 kg Size : 0.6 x 0.3 m Maximum Speed : N/A
CSIRO	Worm Type 2 Legged+6 Electromagnets Motor Actuated	Weight : 45 kg Size : 0.6 x 4.5 m Maximum Speed: 5cm/sec
WCR by T. Fukuda, Nagoya Univ.(1992)	Crawler Type Multiple Vacuum Pad on a Belt Belt Driven by Motor	Weight : 45 kg Size : 0.6 x 0.3 m Maximum Speed : 5 cm/sec

In order to satisfy the above design specifications, the methods employed to achieve wall adhesion and the navigation of the wall-climbing robots were investigated in this research. As a preliminary study, the various adhesion and navigation methods of the existing wall-climbing robots were analyzed. Another robot classification scheme, based on the adhesion and navigation methods, is shown in Table III.

Table III. Classification of wall-climbing robots based on adhesion and navigation methods

	Non-legged type		Legged type
	Crawler	Worm	
Spec.	Simple structure Flat surface only no steering	somewhat complicated structure curved surface no steering	complex structure free ranging of curved surface corner crossing possible
Actuation mechanism	Wheel or crawler Speed : 5 cm/sec	straight or articulated movement by cylinder speed : 17cm/sec	cylinder or linear motor - NINJA : motor - ROBUG-II : cylinder speed : 16 cm/sec
Model	WCR	CEIT CSIRO	NINJA ROBUG-II Biped Walking Robot

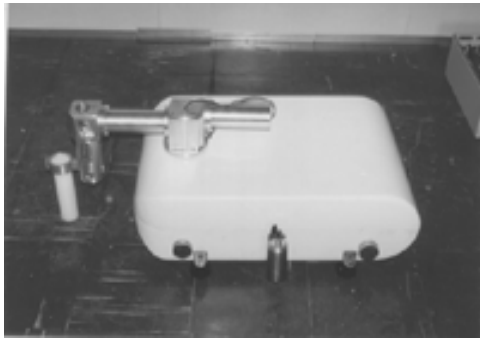
The crawler method is in turn divided into the magnet wheel and vacuum wheel types. The magnet type is not appropriate for this application because a major item to be inspected and decontaminated is the concrete wall. The vacuum-wheel type is also difficult to apply due to the lack of ability to maintain constant velocity movement. Other types, the worm and the legged, consist of multiple complex articulations and a large number of driving motors, so that they are unable to work well under water.

In order to cope with these problems, a new conceptual robot employing a driving mechanism composed of five propellers was devised during this research, as shown in Figure 5. This driving mechanism can control the travelling, traversing, turning, and rotating motions. The dynamic behaviour of the underwater robot and the fluid dynamics introduced by the propeller actions were analyzed and the appropriate control method was selected to control the propeller motion. There were two control methods considered, velocity and torque control. The torque control method took priority over the velocity control method because it had less non-linear characteristics. Therefore, the capacity and the kind of driving motors were selected based on the torque characteristics of the propeller motion.

The manipulator installed in the upper part of the robot is used to carryout the inspection and decontamination. The design of the manipulator is small and lightweight in order to decrease the driving power needed. The structure and weight distribution were also designed to minimize the driving force needed. From this point of view, the various mechanical types were considered, and the SCARA type was finally selected because it is mechanically robust and its motor has a high torque efficiency.

3.4. *Wall-climbing robot*

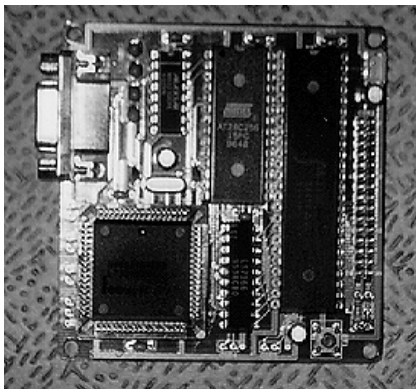
Based on the above design considerations, a wall-climbing robot was fabricated as shown in Figure 5. The robot consists of an underwater navigation module, a manipulator, and a control system.



Prototype



Demonstration robot



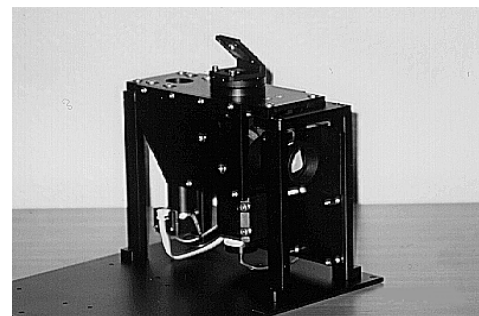
On-Board controller



Servo type propeller



Radiation measuring equipment



Position measuring laser sensor

Figure 5. The underwater wall climbing robot and its components.

The outer structure of the robot is a double skin structure made from Fibre Reinforced Plastic (FRP). The outer surface of robot is double walled to prevent in-leakage of water. Rubber O-rings are installed at the joints and penetrations such as between top and bottom structures of the robot, manipulator attachment, cable, and motor axes. Silicon is applied for extra protection from the water. To provide good heat transfer to cool down the interior electric circuit the bottom plate of robot is made of brass. An anti-desiccant is also located inside the main body to remove vapour.

The robot's movement is controlled by five propellers installed on the robot.

The SCARA type manipulator has two degrees of freedom. The manipulator arms are fabricated from duraluminium, and the power transmission elements are fabricated from tungsten. A radiation detector and a decontamination brush can be remotely attached to the working end of the manipulator.

4. Assessment of radionuclide transport at the research reactor

The objective of this study is the analysis of the impact of residual radionuclides on the area around the TRIGA reactor site after the decommissioning of the TRIGA reactor. Streams, valleys, ridges and water table in the study area were investigated to establish the baseline conditions. The soil in the study area was sampled and its hydraulic parameters were measured. The impact of radioactivity on the area around the TRIGA reactor after 5, 10, 20, and 30 years was analyzed using a 3 dimensional numerical model. The groundwater flow velocity, calculated with MODFLOW, was used as input data. It was assumed that the major residual radionuclides at the TRIGA reactor site were ^{60}Co , ^{137}Cs , and, ^{90}Sr with an average concentration of 1.0 following decommissioning. The boundaries used in the modelling are as follows: the north boundary is the stream in the Barrae valley; the south boundary is the Sinnæ stream; the east is a mountain ridge 80–100 m above sea level; and the west is a line at longitude 127° 04' 31". The area inside these boundaries is named the study area and contains 3.8 km². The study area was divided into 4 layers with the following thicknesses. The upper side of the 3 m thick 1st layer is the water table. The thickness of the 2nd, and 3rd layers are 7 m, and 20 m, while the thickness of the 4th layer varies between 30 and 100 m. The bottom side of the 4th layer is 38 m below sea level (Figure 6).

Groundwater flow modelling was created from MODFLOW (A Modular three dimensional finite difference groundwater flow modelling program). The governing equation

$$\frac{\partial}{\partial x} \left(K_x \frac{\partial h}{\partial x} \right) + \frac{\partial}{\partial y} \left(K_y \frac{\partial h}{\partial y} \right) + \frac{\partial}{\partial z} \left(K_z \frac{\partial h}{\partial z} \right) = S_s \frac{\partial h}{\partial t}$$

for MODFLOW is:

Where:

K_x , K_y and K_z represent the hydraulic conductivity in the x, y and z dimensions (L/T);
 h is the hydraulic head(L);
 S_s is the specific storage(1/L); and
 t is time(T).

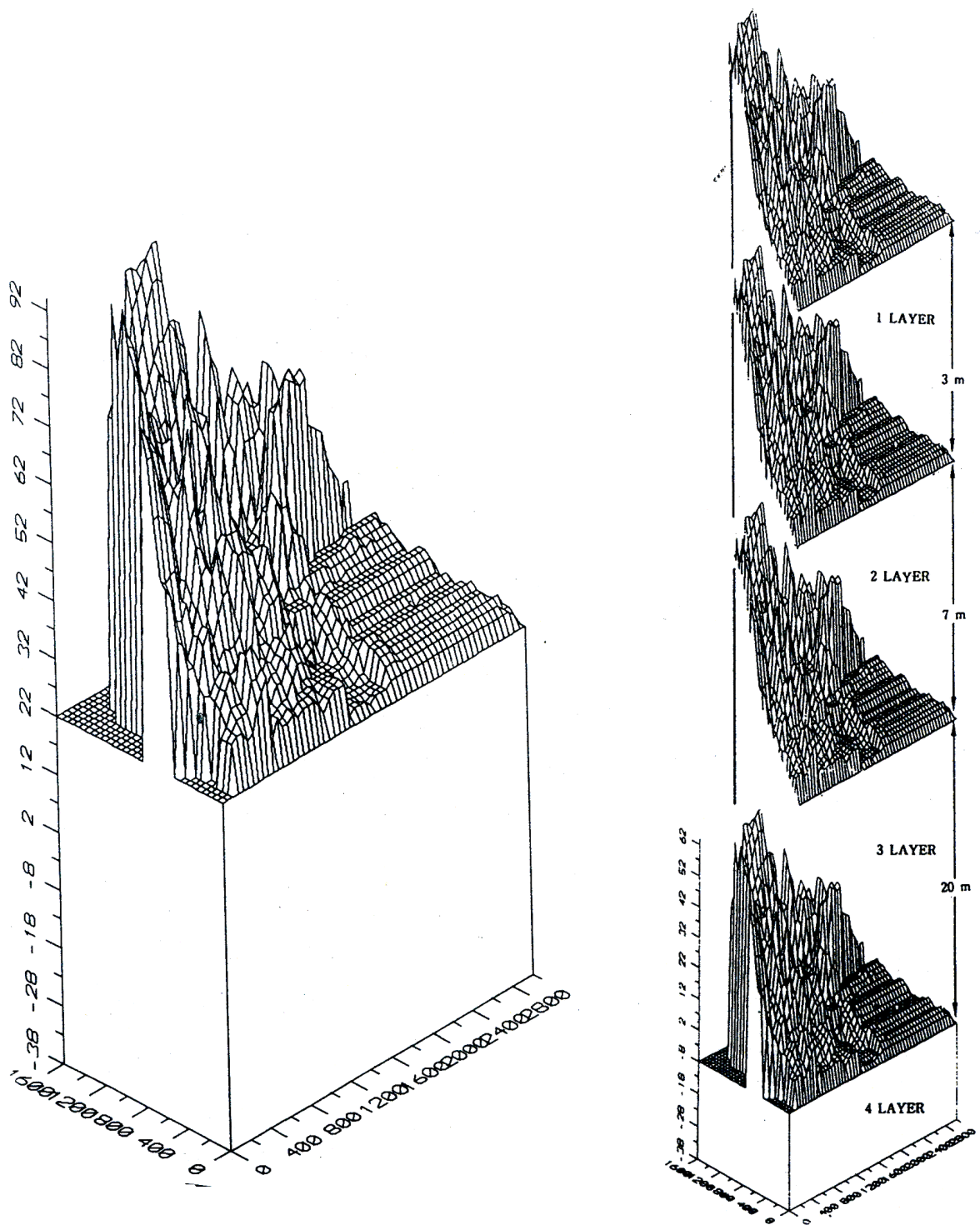


Figure 6. Configuration of the water table and 4 layers.

The computer code MT3D (A Modular Mass Transport 3-Dimension) was used to model the radionuclides. The governing equation for MT3D is:

$$\frac{\partial C}{\partial t} = D \frac{\partial^2 C}{\partial x^2} - v \frac{\partial C}{\partial x} + \frac{\rho_b}{n} \frac{\partial S}{\partial t} - \lambda \left(C + \frac{\rho_b}{n} S \right)$$

$$S = KdC$$

$$D = \alpha v + D^*$$

Where: ρ_b is the bulk density(M/L³); C is the contaminant concentration (M/L³); S the absorption concentration(M/M); t the time(T); v is the flow velocity (L/T); D is the dispersion coefficient(L²/T); λ the material decay constant(T⁻¹); D* the molecular diffusion coefficient(L²/T); and T is the longitudinal or transverse dispersivity (L).

Computation of the water table level was based on the elevation above sea level of the stream surfaces in the study area. The water table at other points was then interpolated from the stream data.

A finite difference net was prepared by dividing the study area, which is 3 × 1.6 km in area, into cells of 50 × 50 m. The number of finite differences cells in a given plane is thus 60 × 32 = 1920. As the study area consists of 4 layers, the total number of finite differences cells is 7680 (Figure 7). Assuming that the TRIGA reactor area was contaminated by ⁶⁰Co at an average concentration of 1.0 following decommissioning, then the average concentration at the reactor area after 10 years is 0.3, and the average concentration 15m distance from the TRIGA reactor boundary after 10 years is 0.003 (Figure 8).

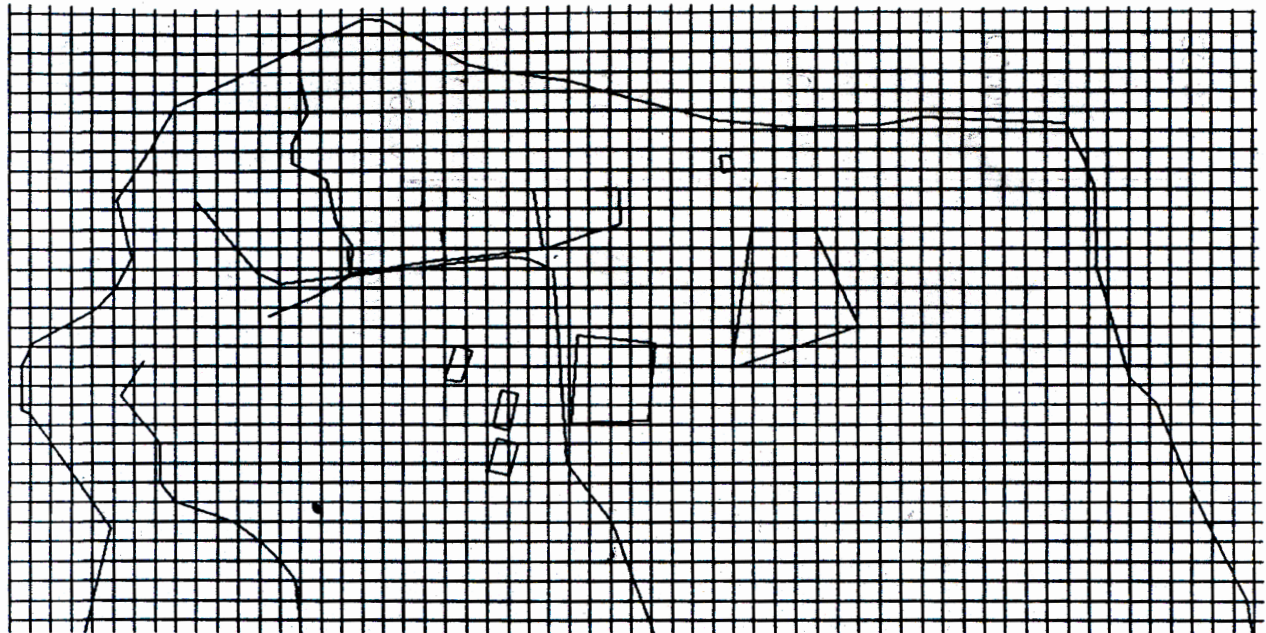


Figure 7. Finite element net of the study area.

5. Soil decontamination

5.1. Soil washing

The objective of the soil washing study is the development of decontamination technology to be applied to soil stored in KAERI which has a radioactivity content below 0.4 Bq/g. The scope of the study includes the collection, drying, sieving and the use of XRD analysis to investigate the characterization of the soil (Figure 9) and laboratory scale experiments to study the sorption and desorption characteristics of cobalt ions with EDTA (Figure 10) or citric acid under various solution pHs. Various liquid waste decontamination treatment methods were also investigated. The design and fabrication of soil washing equipment, chemical makeup systems and batch type decontamination reactors were investigated as was the drying of soil on a conveyor system (Figure 11).

Experimental results indicated that the soil contains Hematite and Wustite, iron oxides which are dissolved during decontamination. Since the amount of desorbed cobalt ions is affected by the dissolved iron ions, the control of iron ions in solution is important in reducing the radioactive waste volume (Figure 12).

5.2. Solvent flushing

This study was undertaken to develop technology for an in situ cleanup process that can be used on a nuclear site to be used in the event of unexpected contamination or a nuclear accident. The work scope included the design and fabrication of laboratory scale solvent flushing equipment (Figure 13), and model development applicable to explaining the decontamination characteristic of soil contaminated with Sr^{2+} ions. Collection, drying, sieving and analysis were executed for the characterization of the soil. For the decontamination test, citric acid was used as a decontamination agent at 25°C, and a given quantity of effluent was collected. The Sr^{2+} ion concentration in the effluent was analyzed by atomic absorption spectroscopy.

A non-equilibrium sorption solute transport code was written in FORTRAN 77 using the Galerkin finite element method with a Linear Basis Function. The matrix calculations used the Thomas algorithm. Time differentiation used the Implicit Difference Scheme

$$\frac{\partial C}{\partial t} + F \frac{\rho}{\theta} \frac{\partial S_1}{\partial t} + \frac{\rho}{\theta} \frac{\partial S_2}{\partial t} = D \frac{\partial^2 C}{\partial x^2} - v \frac{\partial C}{\partial x} - \lambda \left(C + \frac{\rho}{\theta} S_2 \right)$$

$$S_1 = K_p C$$

$$\frac{\partial S_2}{\partial t} = k_2 ((1 - F) K_p C - S_2)$$

Measured values of the input parameters are as follows. The dry bulk density is 1.55 g/cm³; porosity is 36.27 %; water content is 12.00 %; and the pH is 4.3. The hydraulic conductivity and pore velocity were measured from the accumulated effluent volume through the solvent flushing column, during a period of 90 minutes under 1atm. The measured hydraulic conductivity is 5.2×10^{-4} cm/min, and pore water velocity is 0.103 cm/min. The hydrodynamic dispersion coefficient was measured in the solvent flushing column with uranyl (U_2O) solution. Ogata's analytical solution for one-dimension was calculated to be 1.5 m²/min using Mathematica software. The distribution coefficient is 0.2 L/kg in citric acid solution, and the instantaneous adsorption ratio is 0.1. Also, 92.7 % of the Sr^{2+} ions removed

were extracted in the pore volumes from 1 to 4. Effluent concentration was decreased to 16 % of the initial concentration after 10 pore volumes passed. Meanwhile, numerical values are in good agreement with the experimental ones (Figure 14).

5.3. *Electrokinetic soil decontamination*

The characteristics of electrokinetic soil decontamination are discussed below (see also Figure 15). This method is applicable to heavy metals, organic compounds and radionuclides. It can be used on-site for heavily contaminated areas, and is being developed for use in less heavily contaminated areas. The objective of this study was the development of electrokinetic soil processing technology for soil contaminated with ^{60}Co , ^{137}Cs or ^{90}Sr . The study included the determination of input parameters and the optimization of decontamination efficiency. The work scope covered the design and fabrication of laboratory scale test equipment, the investigation of decontamination efficiency, and the modelling of decontamination behaviour.

The governing equation of electrokinetic remediation is:

$$\frac{\partial nC_i}{\partial t} R d_i = \frac{nD_i}{\tau} \frac{\partial^2 C_i}{\partial x^2} + (u_i z_i F + \frac{\varepsilon \zeta}{\mu}) \frac{\partial \phi}{\partial x} \frac{\partial C_i}{\partial x}$$

where:

- D is the diffusion coefficient;
- u_i is the ionic mobility;
- F is the Faraday constant;
- z is the zeta potential; and
- μ is the viscosity coefficient.

The boundary conditions are:

- $C(0,x) = C_0$ for $0 < x < 20$;
- $C(t, 0) = 0$, and $C(0, 20) = 0$; and
- $\phi(t,0) = 40$.

The input parameters for modelling are given in Table IV. As for solvent flushing, the modelling was done with a computer code written in FORTRAN 77, using the Galerkin finite element method with a linear basis function. Time differentiation is via an implicit difference scheme, and matrix calculations used the Thomas algorithm.

Experimental results indicated that when an acetate buffer was injected into the soil, no precipitate formed in the column due to the restraint of the pH increase. Figure 16 shows a high remediation efficiency for the process. Namely, 21% of the total amount of Sr^{2+} in the column (13.9 mg) was removed after remediation for 0.6 days, 33% (21.9 mg) was removed after 0.9 days, 84% (55.8 mg) was removed after 1.6 days, 92% (61.1mg) was removed after 2.5 days, and 97% (64.4mg) was removed after 3.8 days. At the end of the run, the kaolin clay was drawn out and divided into several pieces. The concentration in the pore solution was measured by atomic absorption spectroscopy. The predicted values of the residual concentration after remediation calculated by the modelling code were fairly similar to the experimental values. In conclusion, 97% of the total amount of Sr^{2+} in the column was removed after remediation with acetate buffer for 3.8 days.

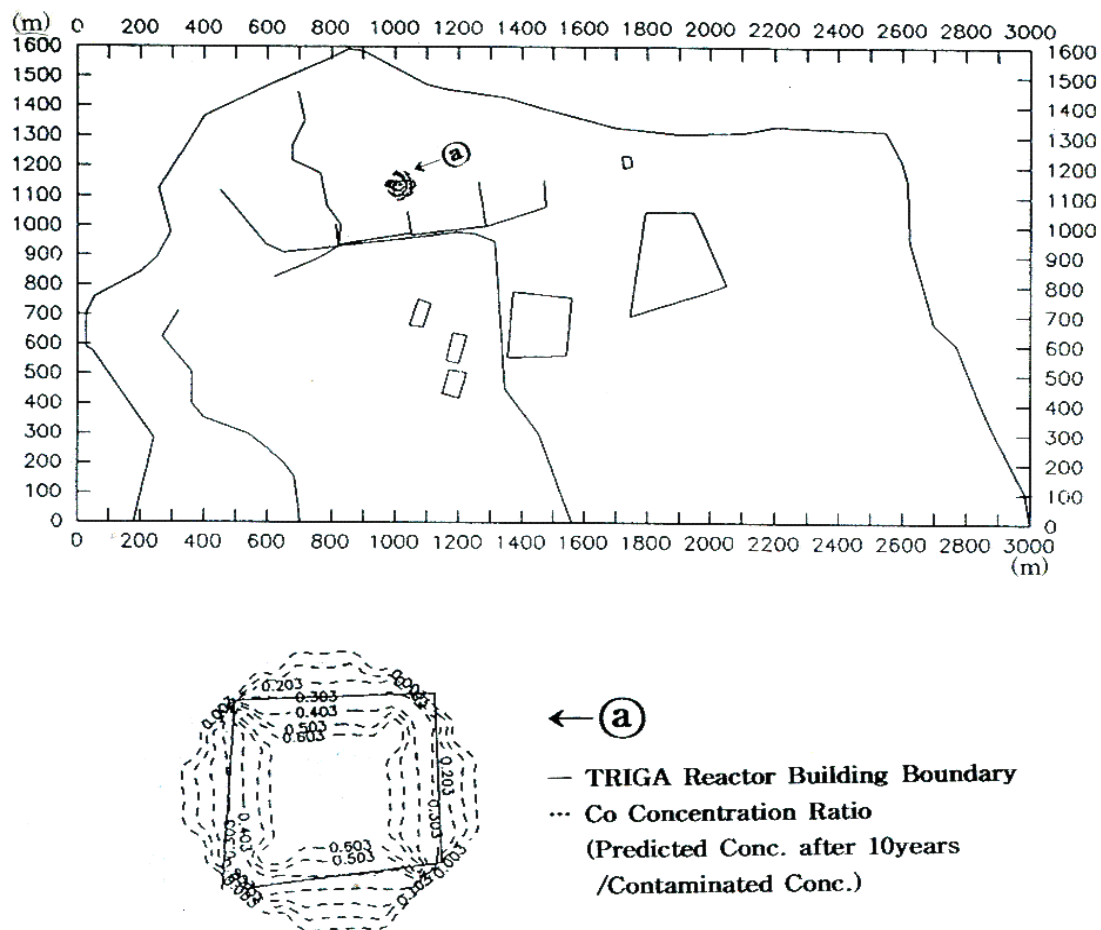


Fig.8. ^{60}Co Transport assessment around TRIGA building after 10 years (TRIGA building area was assumed to be contaminated with 1.0ppm Co).

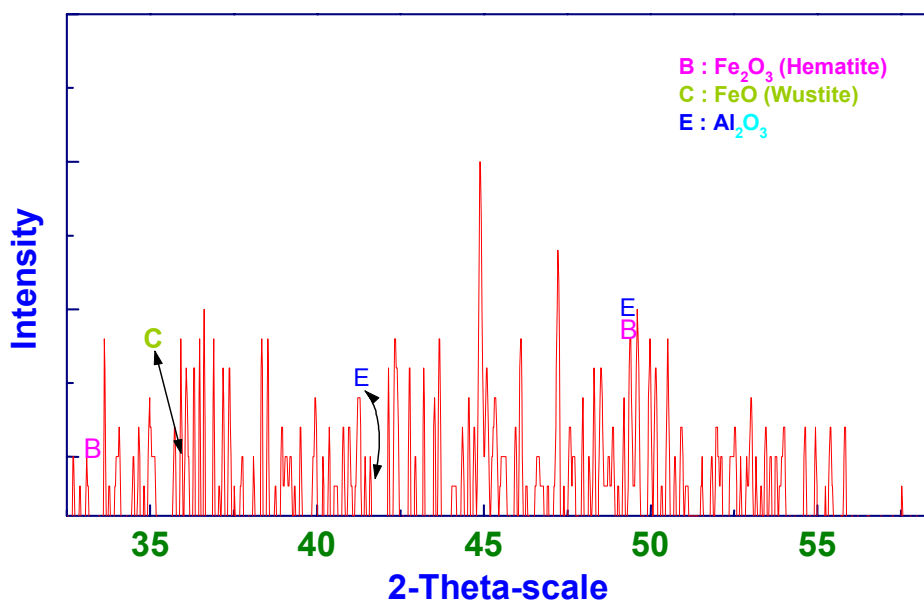


Figure 9. XRD pattern of soil before decontamination.

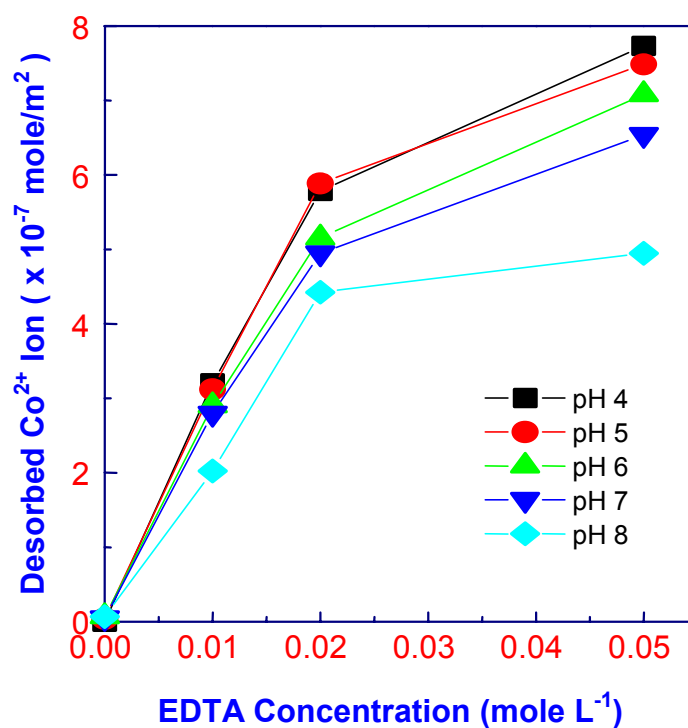


Figure 10. Desorption of Co^{2+} ions according to the $[\text{EDTA}]$ at various solution pH.

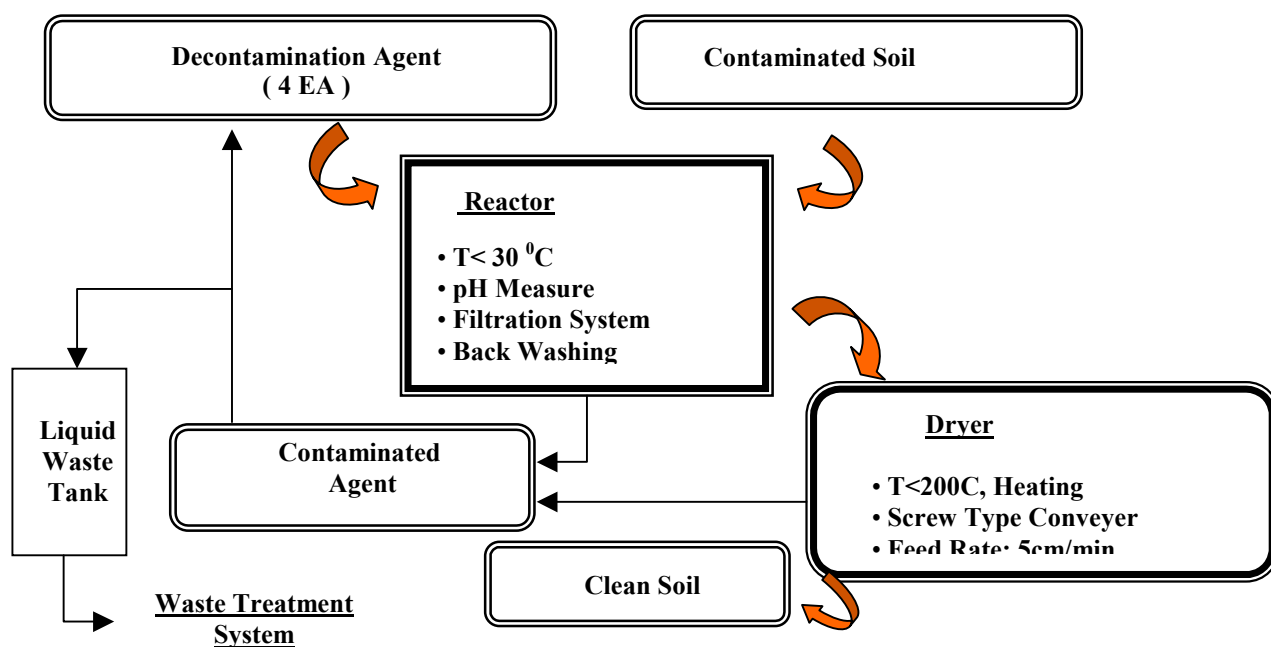


Figure 11. Soil washing equipment.

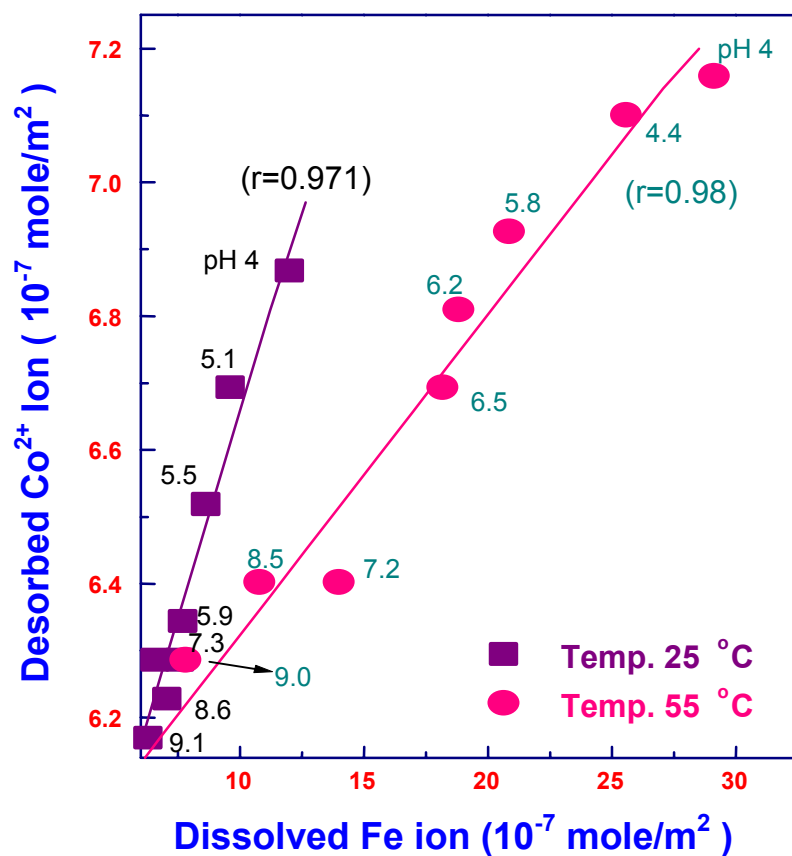


Figure12. Correlation between Fe ion and Co^{2+} ions in EDTA solution.

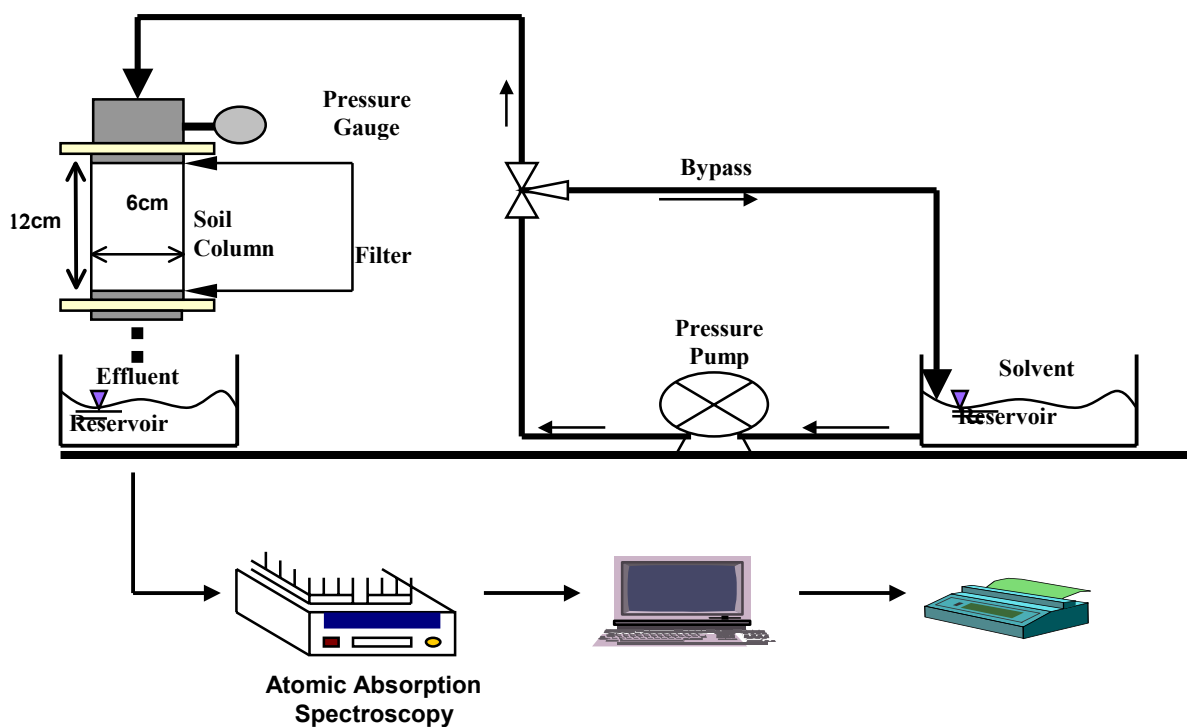


Figure 13. Apparatus for solvent flushing.

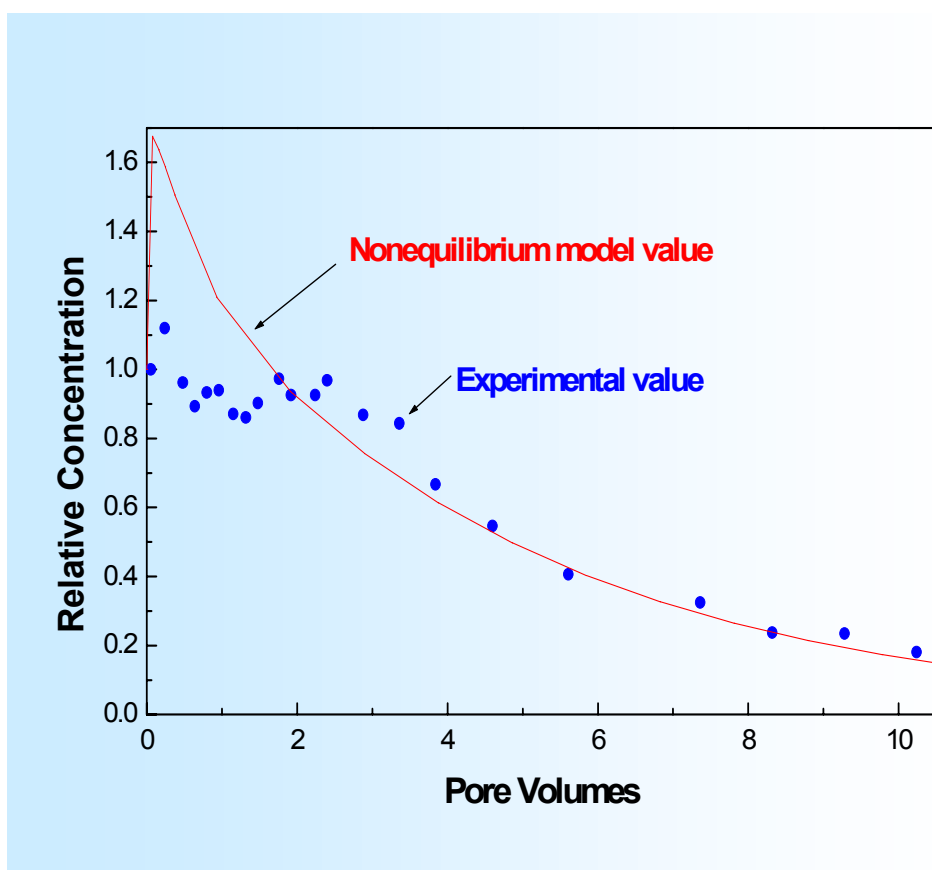


Figure 14. Experimental results of solvent flushing by citric acid solution.

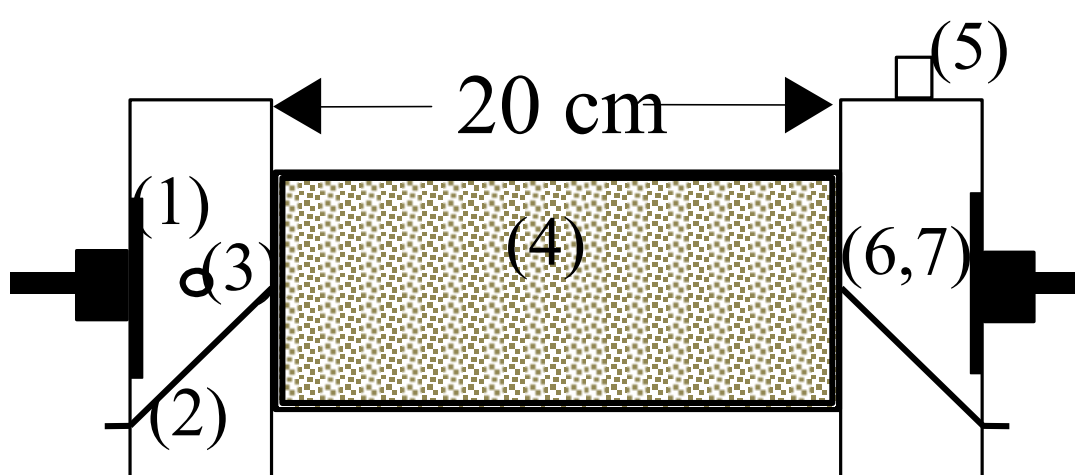


Figure 15. Schematic diagram of electric cell. (1) Ti electrode, (2) Pt wire, (3) effluent out, (4) clay, (5) water in, (6) filter paper, (7) stainless steel sieve.

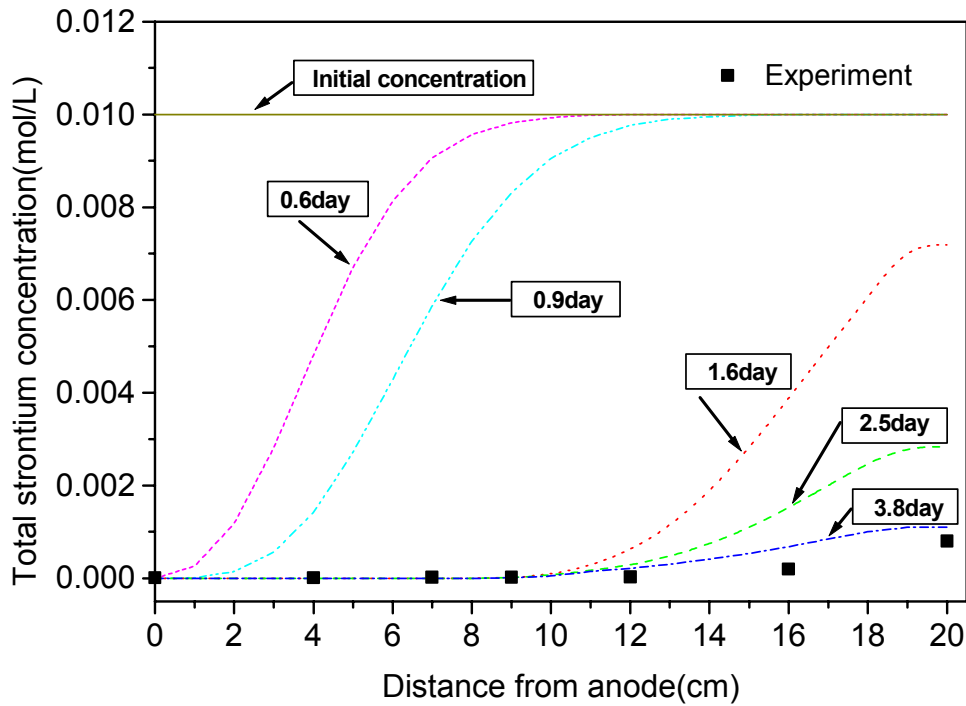


Fig.16. Distribution of total strontium concentration in soil column versus time.

Table IV. Input parameters for electrokinetic modelling

Parameter	Value
Bulk density	0.97 g/cm ³
Cell length	20 cm
Cell diameter	2.8 cm
Temperature(K)	298
Porosity	0.25
Applied potential	40 V
Faraday constant	96487 C/mol
Effective Di	1.5×10^{-2} cm ² /min
Kd	3.11 ml/g
Effective ionic mobility	0.584 cm ² /V. min
Electroosmotic velocity	8.28×10^{-3} cm/min at 40V

6. Conclusions

From KAERI's research and development work on the decommissioning of research reactors and environmental remediation, the following conclusions can be drawn.

A 3-dimensional graphic simulator using IGRIP and its actual application to the TRIGA reactor dismantling process showed satisfactory performance for the investigation of the feasibility of the decommissioning process and for the training of radiation workers. The test results of the underwater wall-ranging radiation inspection robot in the TRIGA reactor pool were satisfactory. They indicated that the robot was especially useful in measuring the radioactive contamination map of the non-metallic surfaces in the water.

From the assessment of the underground radionuclide transport around the TRIGA reactor building, it was found that the radionuclides would have no significant influence on the environment in future due to their slow migration rate, their decay and dilution. The soil decontamination results showed that, by using the soil washing method described, more than 80% of soil wastes could be decontaminated sufficiently to allow them to be discharged to the environment. It was also determined that the control of solution pH and temperature is important for the reduction of decontamination waste in the soil washing process. Further decontamination, however, was found to be needed for the residual soil waste, consisting mostly of fine soil particles, by using the electrokinetic decontamination method.

7. Future research work

The D & D works for the Korean TRIGA reactors will be completed in 2007. Along with the D & D work, soil remediation and residual radioactivity assessment will continue to be studied until 2007. Surface decontamination of metal waste and the treatment of uranium sludge wastewater will also be studied. Meanwhile, development of radiation dose measurement equipment by remote control, graphite treatment research, and remote cutting equipment will be developed in the near future.

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Refurbishment of Pakistan research reactor (PARR-1) for stainless steel lining of the reactor pool

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Abstract. Pakistan Research Reactor-1 (PARR-1) is a pool-type research reactor. Reactor aging has resulted in the increase of water seepage from the concrete walls of the reactor pool. To stop the seepage, it was decided to augment the existing pool walls with an inner lining of stainless steel. This could be achieved only if the pool walls could be accessed unhindered and without excessive radiation doses. For this purpose a partial decommissioning was done by removing all active core components including standard/control fuel elements, reflector elements, beam tubes, thermal shield, core support structure, grid plate and the pool's ceramic tiles, etc. An overall decommissioning program was devised which included procedures specific to each item. This led to the development of a fuel transport cask for transportation, and an interim fuel storage bay for temporary storage of fuel elements (until final disposal). The safety of workers and the environment was ensured by the use of specially designed remote handling tools, appropriate shielding and pre-planned exposure reduction procedures based on the ALARA principle. During the implementation of this program, liquid and solid wastes generated were legally disposed of. It is felt that the experience gained during the refurbishment of PARR-1 to install the stainless steel liner will prove useful and better planning and execution for the future decommissioning of PARR-1, in particular, and for other research reactors like PARR-2 (27 kW MNSR), in general. Furthermore, due to the worldwide activities on decommissioning, especially those communicated through the IAEA CRP on "Decommissioning Techniques for Research Reactors," the importance of early planning has been well recognized. This has made possible the implementation of some early steps like better record keeping, rehiring of trained manpower, and creation of interim and final waste storage.

1. Introduction

The Pakistan Research Reactor-1 (PARR-1) is a pool-type research reactor originally designed to operate, with 93% Highly Enriched Uranium (HEU) fuel, at a maximum power level of 5 MW(th). It rendered invaluable services in the training of manpower, production of radioisotopes and basic research for about 25 years. Due to proliferation resistant policies initiated during late 80s, the HEU fuel was no longer available. Thus it was decided to convert the reactor core to the currently available 19.9% Low Enriched Uranium (LEU) fuel and raise the neutron flux by upgrading the reactor power to 10 MW(th). This conversion and upgrading could only be accommodated by making some adjustments in the reactor systems, i.e. the primary cooling system (to dissipate the 5 MW(th) additional power), which in turn requires the partial decommissioning of PARR-1 before beginning adjustments/improvements in the facility and installation of equipment for the core conversion and upgrading. It was a major activity involving the reactor, its support systems, primary cooling system, process instrumentation and reactor pool. If only the power was being upgraded, this activity could have been limited to the primary cooling system and process instrumentation but due to the aging related increase of seepage from the pool walls, it was decided to augment the pool walls with a stainless steel liner. For this purpose, decommissioning activities required unhindered access to the pool.

Extreme care was exercised as high radiation fields were involved. Decommissioning procedures were prepared for dismantling of major components. Measures were taken to protect the personnel. However, unavoidable exposures were planned according to the ALARA principle. After the commissioning of the interim fuel storage bay and a cooling period of about three months for fuel and active reactor components, dismantling of the core was begun.

All active core components including the fuel elements and graphite reflector elements were removed from the reactor pool and transferred to an interim storage bay via a fuel transfer cask. The core support structure with the grid plate was completely removed from the reactor bridge and stored temporarily in a specially built shielded arrangement. The ceramic tiles from the reactor pool as well as from the holdup tank were decontaminated and lined with a stainless steel lining. Major modifications were made to the cooling system that included:

- (i) replacing the aluminium primary piping with stainless steel,
- (ii) installing a new set of pumps,
- (iii) adding two more heat exchangers, and
- (iv) enhancing the cooling capacity of the cooling tower.

The old piping was decontaminated and made available for re-use.

2. Description of PARR-1

PARR-1 (originally operating at 5 MW(th)) is a pool-type research reactor. It is cooled, moderated, reflected, and shielded by demineralized water. Overall shielding is provided by light water and high-density concrete. The reactor core is immersed in a reactor pool having two sections, called the open and stall pools. The cross-sectional view of the reactor is shown in Figure 1. A concrete wall containing a tapered opening that could be opened or closed with a removable watertight aluminium gate separates the two sections. The stall pool contained all the beam tubes, pneumatic rabbit terminals and the graphite thermal column. The open pool has a large area for bulk irradiation, a transfer port for underwater transfer of irradiated samples to the hot cell and a gamma cell for dry gamma irradiation. An aluminium tower supporting the reactor core is suspended from the manually operated bridge. One Pu-Be neutron source (1×10^7 n/sec) is placed near the core.

Heat generated in the core is dissipated in water by natural convection at low power levels and through forced circulation of water at higher power levels. The primary cooling water flows by gravity downwards through the reactor core, grid plate and plenum into the holdup tank. Subsequently the water is drawn from holdup tank by the main circulating pumps through the shell side of the heat exchangers and back in the pool as shown in Figure 2.

The experimental facilities consist of 6 radial beam tubes, a tangential through tube, graphite thermal column, three independent pneumatic rabbit systems, a hot cell, dry gamma irradiation cell, and a bulk irradiation area. These facilities are depicted in Figure 3.

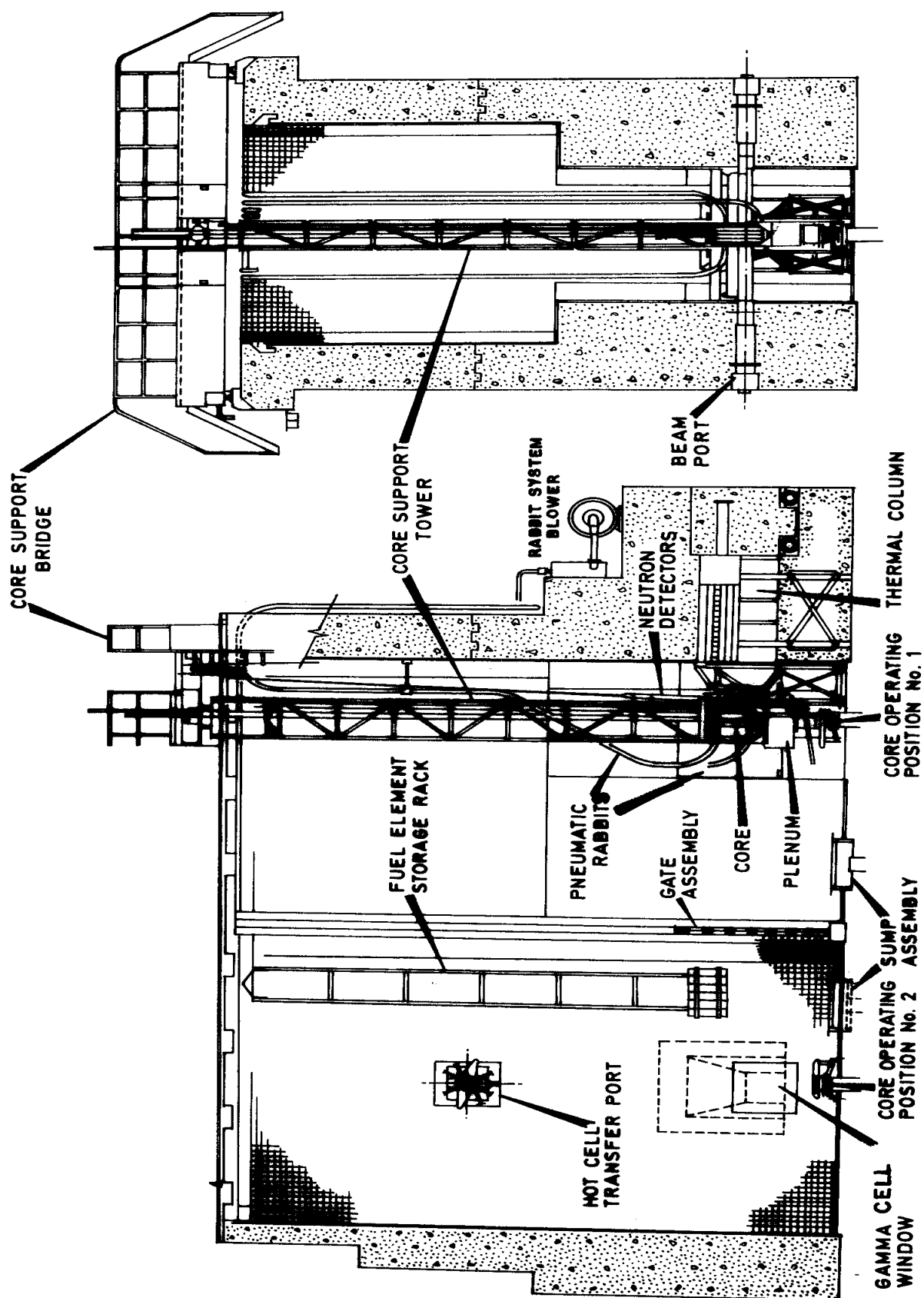


Figure 1. Vertical sections through reactor.

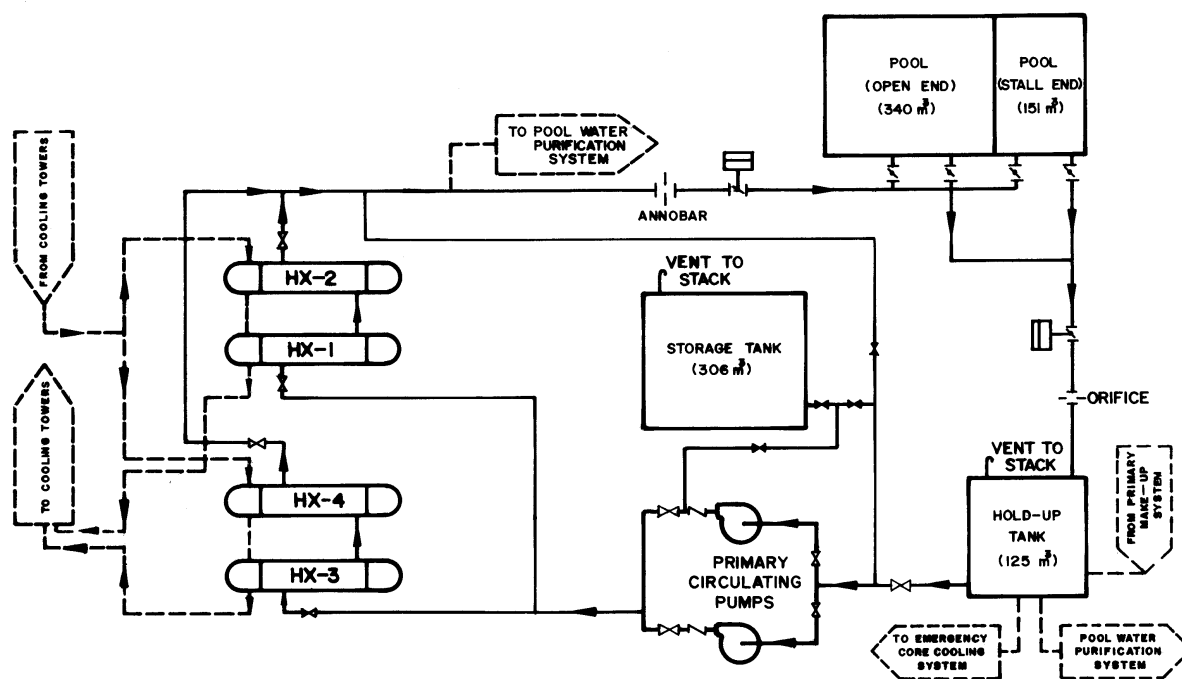


Figure 2. Primary cooling system.

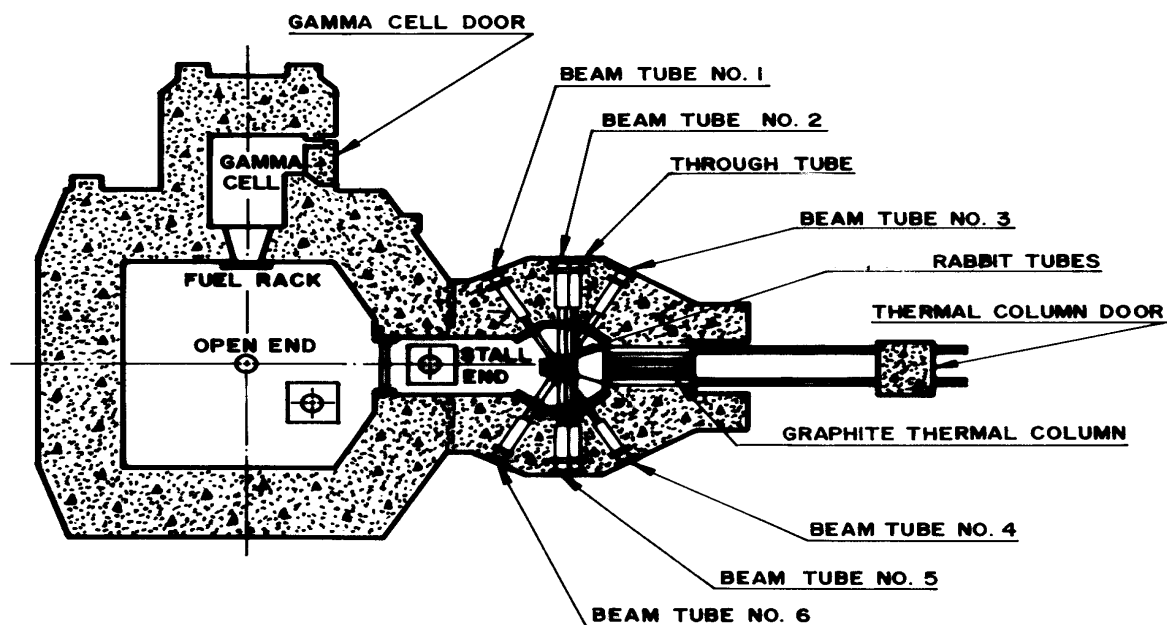


Figure 3. PARR-1 experimental facilities layout.

3. Preparation for decommissioning

Preparations made before the decommissioning included:

- (i) relevant information gathering, commissioning of interim spent fuel storage bay, activity assessment of irradiated components, fabrication of fuel transfer cask, procurement/fabrication of equipment/tools, and documentation.

3.1. Relevant information gathering

At this stage of the decommissioning a coordinated effort (between all those who were involved) was made to gather relevant information from open sources, e.g., IAEA, US NRC. In this process, the gathered information was reviewed again and again till it is decided by consensus that sufficient information is there.

3.2. Spent fuel storage bay

In order to make the reactor pool area accessible, all the reactor components had to be removed and stored in a separate storage area. Therefore, an interim spent fuel storage bay was constructed and filled with demineralized water.

The interim spent fuel storage bay comprises four underground, interconnected pools made from reinforced concrete and lined with stainless steel (Figure 4). It has been designed to store 400 irradiated fuel and reflector elements.

Suitable arrangements have been provided in the interim spent fuel storage bay for:

- (i) handling of spent fuel transfer cask and spent fuel element, ventilation, water purification, radiation monitoring, and other safety requirements.

3.3. Activity assessment

Before dismantling the reactor systems, some of the reactor components (not core components) were surveyed for radiation/contamination. The dose rates from the active components that were measured were helpful in planning and executing the handling of these components.

3.3.1. Analysis of pool water

Samples from the pool water were analyzed. The activity was measured and found to be less than 1.5 Bq/ml (4×10^{-5} μ Ci/ml), which was due to Ag-110m (no other source was found).

3.3.2. Analysis of pool tiles

Samples of the ceramic tiles lining the pool were analyzed for contamination/radiation. Tiles, which were not exposed directly, were analyzed and the radioactivity level was found to be less than 0.04 Bq/g (1×10^{-6} μ Ci/g) due to Ag-110m, Bi-214 and Pb-214. This radioactivity was found to be slightly above the background level. However ceramic tiles on the floor of stall end near the core outlet were found to have a maximum dose rate of about 1 mSv/h.

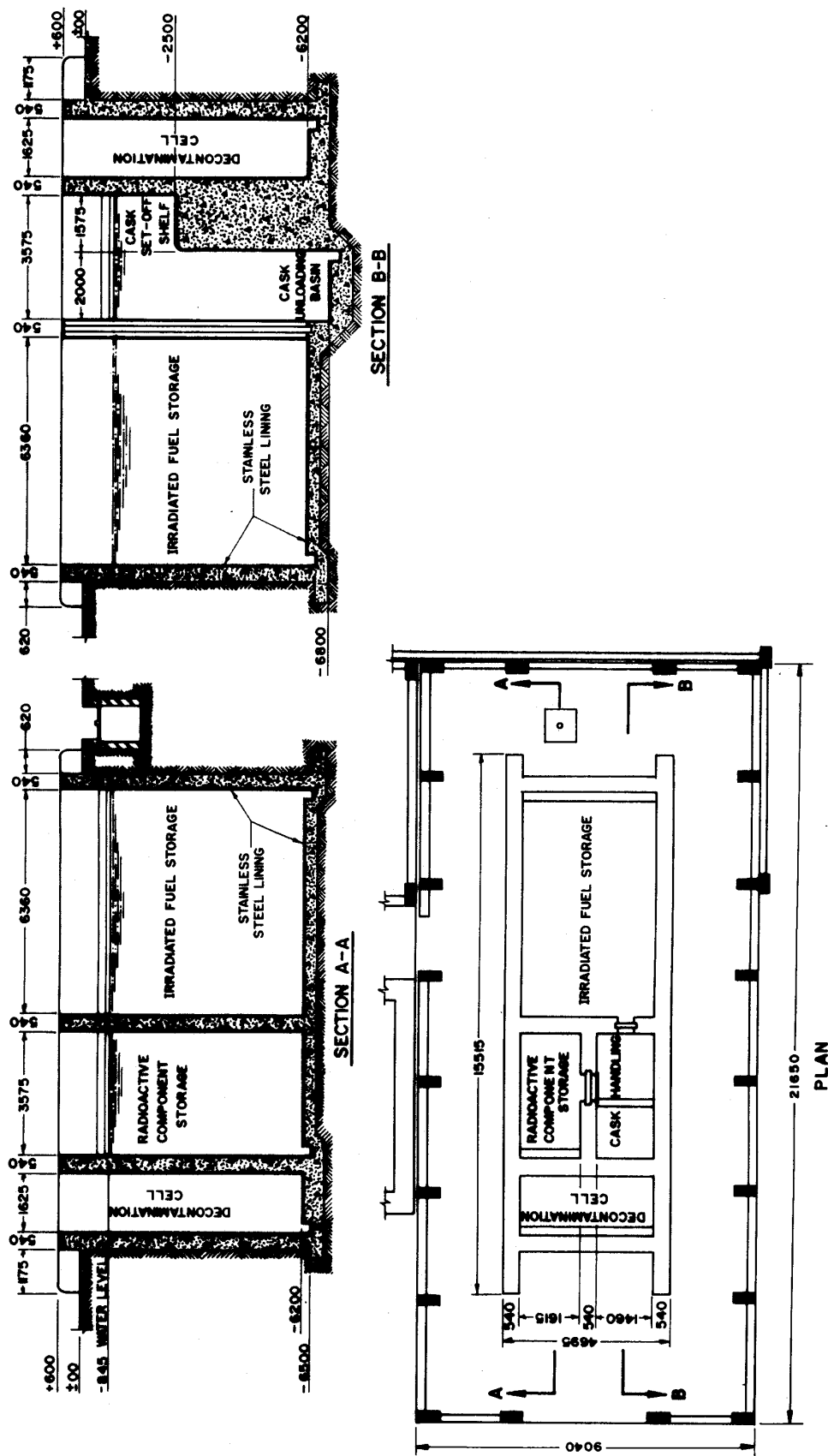


Figure 4. Spent fuel storage bay.

3.3.3. Dose assessment of reactor components

Dose rates were assessed at the end of the beam tubes, pneumatic rabbit tubes, the extension of thermal column, thermal shield, grid plate and graphite reflector elements. The dose rates are given in Table I.

Table I. Activity assessment of core components

No.	Components	Dose rate (mSv/h)
1.	End of beam tubes	5 to 10
2.	End of pneumatic rabbit tubes	10
3.	Thermal column extension	100
4.	Thermal shield	Nil
5.	Grid plate	400
6.	Reactor core support structure	8
7.	Plenum	340
8.	Graphite reflector elements	5 to 400

3.4. Fuel transfer cask

A fuel transfer cask was designed and fabricated (Figure 5) in accordance with the following IAEA regulations for the safe transportation of radioactive materials:

- The radiation level originating from the package shall not exceed 2 mSv/h (200mr/h) at any location on the external surface of the package during normal transport; and
- The transport index at any time during normal transfer shall not exceed 10.

The cask was conveniently handled in the reactor hall as well as in the interim fuel storage bay utilizing the overhead cranes at the respective locations. The lead shield thickness of the cask has proven sufficient to meet the radiation shielding requirements of four irradiated fuel elements clustered together.

3.5. Procurement/Fabrication of Equipment/Tools

Numerous tools were either acquired or designed/fabricated locally for the handling of the active components. Almost all of these were used for the remote and underwater handling of active components; however, some of these were used for the handling of components like beam-tubes, etc.

3.6. Documentation

There has been a deliberate effort to ensure that the whole process of partial decommissioning should be documented in an appropriate format such that it can be readily accessed and interpreted for any possible future use.

4. Partial decommissioning of PARR-1

Decommissioning procedures were adopted and then written, from relevant IAEA guidelines and available US NRC documentation, based on the scope and magnitude of modifications/changes involved. Consequently these procedures received approval from the National Regulatory Authority. Checklists were created and completed during each decommissioning activity. A sample of such a checklist for beam tube decommissioning is given in Appendix A.

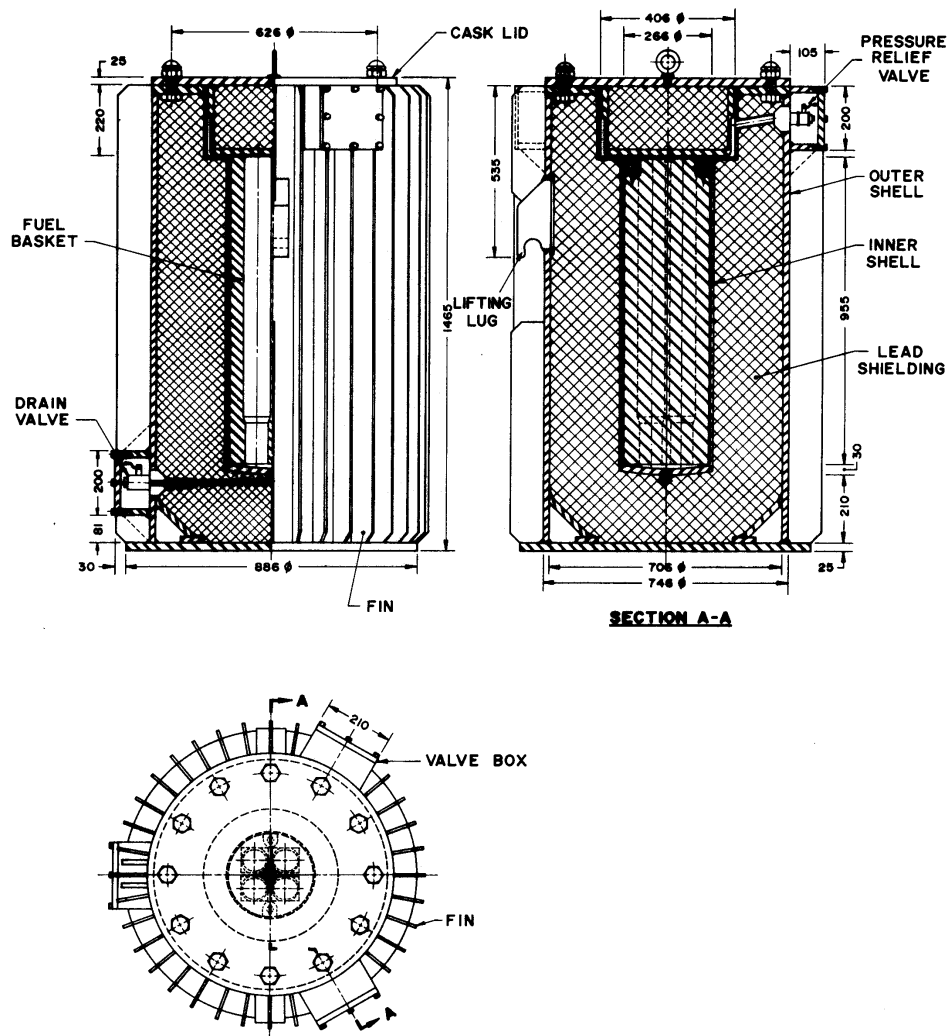


Figure 5. Irradiated fuel transfer cask.

Some of the decommissioning activities are described below.

4.1. Beam tubes

All the beam tubes are identical; consequently the decommissioning procedures were also identical. For illustration purpose, the procedure for beam tube No. 1 is described in the following:

- Removal of beam tube No. 1 was carried out in accordance with the written procedure (Appendix A).
- Area radiation monitoring was conducted.
- Lead and concrete plugs were removed using special tools.
- The reactor bridge was moved to the open end of the pool and the pool-dividing gate was installed to allow the draining of the stall end to below the beam tube level
- The water was stored in the holdup and storage tanks
- Bolts holding the tube were removed
- The tube was taken out, the measured dose rates were found to be up to 40 mSv/h

- The tube was wrapped in polyethylene sheet and stored in the reactor hall at storage locations mentioned in Table II
- The surface dose rate at the containment wall (reactor hall) was found to be 50 $\mu\text{Sv/h}$.

Similarly, the remaining beam tubes were removed and stored accordingly.

Table II. Storage location of beam tubes

Beam Tube Number	Storage Location	Maximum Dose (mSv/h)	Contact	Storage Dose ($\mu\text{Sv/h}$)	Surface
1	Beam port floor N-5	40		50	
2	Beam port floor N-7	100		100	
3	Gamma Cell	18		-	
4	Gamma Cell	55		-	
5	Beam-port floor N-6	100		70	
6	Beam-port floor N-8	-		60	

4.2. Thermal shield

The thermal shield was removed using the following procedure:

- the reactor bridge was moved to the open end and the pool dividing gate was installed;
- the reactor pool was drained below the beam-tube level;
- a radiation survey was carried out in the stall end of the pool; and
- the thermal shields were removed, properly cleaned, washed, wrapped, and stored

4.3. Reactor core components

- A total of 65 standard, 17 control, 2 partial and 9 reflector elements were transferred to the storage bay
- The dose rate at the driver's seat of the transporting vehicle was less than 20 $\mu\text{Sv/h}$

4.4. Neutron source

A special wax-shielded container was used for transferring the neutron source (Pu-Be with a strength of 1×10^7 n/sec) to the storage bay. The source was placed in the container under water and was stored in the component section of the bay.

4.5. Thermal column

Because of the high radiation dose on the lead plate, identification of the lifting points became very difficult. The column was eventually removed according to established procedures (similar to those described earlier in Section 4) and stored on the beam port floor in a shielded area.

4.6. Core support tower

The contact dose inside the holes of the grid plate varied from 0.5 to 2.3 Sv/h and over the surface the dose rate was about 0.4 Sv/h. Due to its large size, high dose rate and alignment problems, removal of the tower was planned very carefully. The dose rate was determined and a special shielding arrangement was made in the reactor hall for temporary

storage of the core support structure. The tower was suspended vertically on the beam port floor, with the lower active part of the tower properly shielded. Following the conduction of a radiation survey the area was cordoned off.

4.7. Cooling system

Because of the design requirement for upgrading, the entire primary cooling system was dismantled, except for the embedded portion. In the pump room all the piping, valves and pumps except the heat exchangers were dismantled. A plasma-arc cutting machine was used for pipe cutting. As the piping of the secondary cooling system was to be reused in the upgraded design it was partially dismantled in the pump room and near the cooling tower. Several valves and pumps were overhauled, packed and stored in the pump room. Only contaminated aluminium piping was placed in a separate temporary storage area.

4.8. Miscellaneous equipment

The pneumatic rabbit tubes and their supports were removed and stored in a separate storage room in the reactor hall. As the capsules used for irradiation in the rabbit tubes were radioactive, having dose rates of about 0.4 Sv/h, they were stored in a shielded hole on the beam port floor. Fuel element handling tools were stored in a separate storage room. Components of the transfer port were also placed in that room. Storage locations of various other components were recorded.

5. Radiation protection

During the decommissioning, adequate measures were taken to protect the personnel working around the reactor, the general public and the environment against radiation exposure. All the workers were provided with TLDs, dosimeters, dungarees (overall), overshoes, gloves and masks as required. Additional ventilation was provided in the working areas. On-site radiological monitoring and periodical radiation surveys were carried out. In case of unavoidable exposures, the ALARA principle was followed. After completion of daily work, personnel were monitored for contamination and radiation dose. During the period of about ten months from decommissioning to re-startup of PARR-1, the maximum external dose received by a worker was 4.7 mSv (470 mR), which is much below the maximum permissible annual limit. Moreover, whole body counting of the workers heavily involved in the decommissioning activities was arranged and no internal contamination was found.

6. Contamination control and waste disposal

During the decommissioning activities, measures were taken to control contamination. Liquid and solid wastes generated during these activities were treated and disposed of.

6.1. Solid waste

Solid waste generated from PARR-1, consisted mainly of the ceramic tiles removed for making channels to install the base plates and strips on the walls of pool and holdup tank to which the stainless steel liner was attached. The ceramic tiles were decontaminated in situ by flushing water on the pool wall interior before being removed. When working in the contaminated area, it was mandatory for the workers to wear protective clothing. After completion of assigned work, the workers were checked for contamination. The debris produced during the work was put in the containers, which were unloaded into drums placed over polyethylene sheets near the reactor pool. Direct dose from the tiles was negligible. However tiles on the stall pool floor, which had a high dose rate because of the direct

irradiation of these tiles, were kept separated from the other tiles. Analysis of these tiles showed the presence of following isotopes:

^{152}Eu , ^{154}Eu , ^{60}Co , ^{66}Zn , ^{47}Ca , ^{46}Sc and ^{108}Ag .

The total activity was about 2×10^4 Bq/g (6×10^{-5} mCi/g) while other tiles, which were not exposed directly, had only $^{110\text{m}}\text{Ag}$ with activity of the order of 4×10^{-2} Bq/g (10^{-6} $\mu\text{Ci/g}$).

The waste was packed in about 66 drums having a total volume of about 14 m^3 and was disposed of according to instructions of the Waste Management Group of the institute. The number of drums and their surface dose rates are given in Table-III:

Table III. Characterization of solid waste

S.No.	Number of Drums	Dose Range ($\mu\text{Sv/h}$)
1.	5	100 to 360
2.	16	10 to 100
3.	45	<10

Another source of solid waste was aluminium piping discarded from the primary cooling system. These were washed for the removal of loose contamination, wrapped in polyethylene sheet and temporarily placed in a separate store. The pipes are not active but had only negligible fixed contamination. These pipes will be salvaged for any future use in radioactive drainage system.

6.2. Liquid waste

The major source of liquid waste was the primary cooling water in the pool and holdup tank. After a delay time of about three months, the analysis of a sample from the pool water indicated an activity of less than 1.5 Bq/ml (4×10^{-5} $\mu\text{Ci/ml}$) due to Ag-110m. No evidence of other radioactive elements was found. This activity was in the low range, as such, the primary water was sent to the seepage pit in two batches after dilution. The total quantity of liquid waste was estimated at 600 m^3 .

7. Planning for PARR-1 decommissioning

After the refurbishment of the PARR-1 including its conversion from HEU to LEU fuel and upgrading from 5 MW(th) to 10 MW(th), it was concluded, after careful studies that PARR-1 will not be decommissioned within the next 20 to 30 years. No need is felt by the authorities to set up a special group to start the planning for decommissioning at this stage. However, due to the worldwide activities on decommissioning and especially due to this IAEA CRP on “Decommissioning Techniques for Research Reactors”, the importance of early planning has been well recognized. Taking this into account the following steps have been taken:

- the addition of a chapter on decommissioning in the safety analysis report of PARR-1;
- the consideration of different approaches to decommissioning;
- better classification of records and record keeping;

- rehiring of trained staff;
- the addition of an interim spent fuel storage bay of ample capacity
- enhancement of waste disposal facilities
- a proposal for a national repository for disposal of nuclear waste

7.1. Safety analysis report

A special chapter on decommissioning has been added to the final safety analysis report of PARR-1. In this chapter, which was not present in the original document, it is stated that the Pakistan Atomic Energy Commission (PAEC), a state organization, and owner of PARR-1 will be responsible for the allocation of funds for decommissioning. The major considerations in the preparation of the decommissioning plan include:

- the availability of waste storage or disposable facilities;
- the required regulatory criteria before or during the development of the decommissioning plan
- release criteria for radioactive waste
- the establishing of criteria for unrestricted use of the facility by the national regulatory body;
- adequate funds; and
- trained manpower, if necessary the training of personnel on mock-ups to reduce the time spent in radioactive zones.

An organizational chart for decommissioning program is shown in Figure 6.

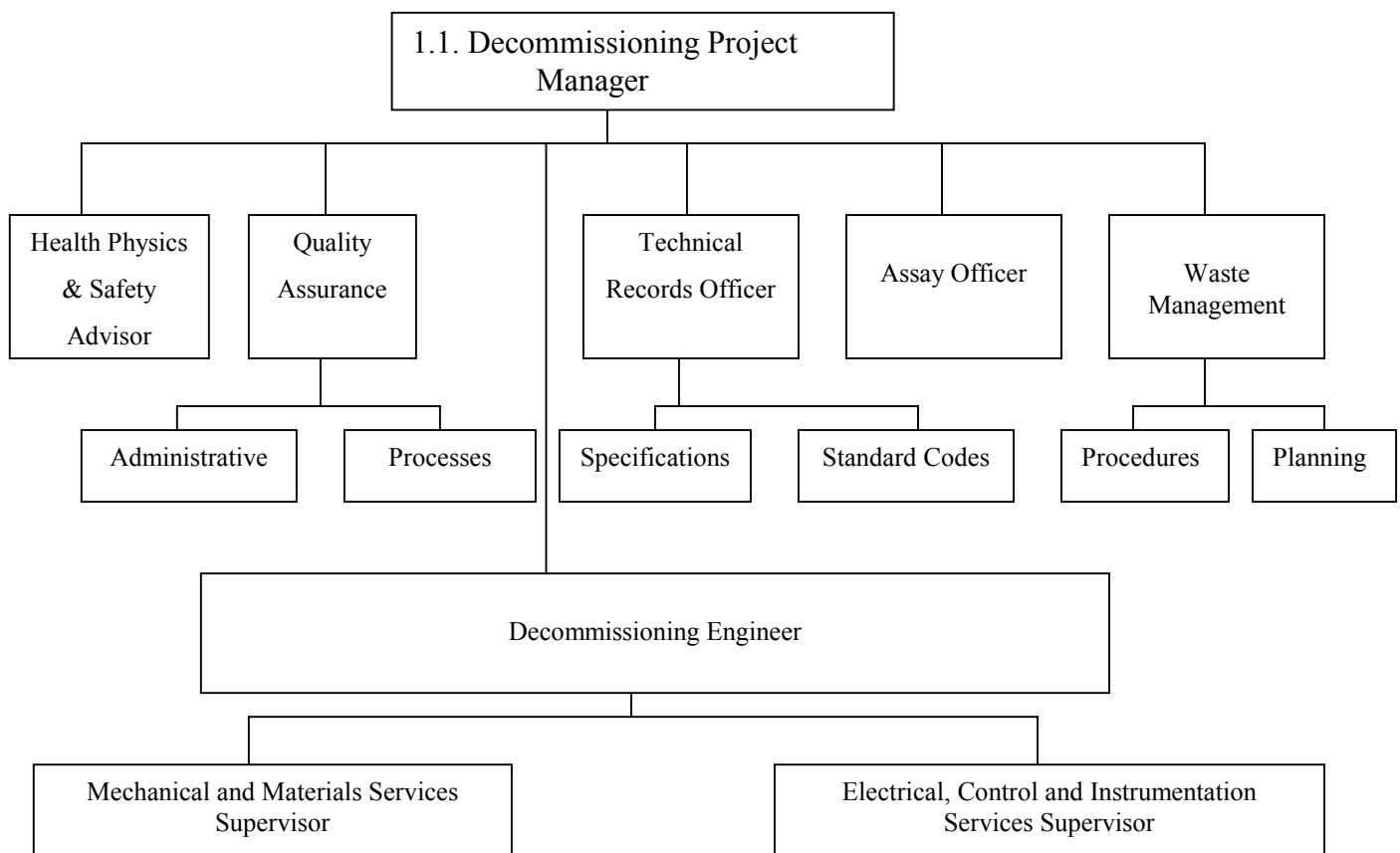


Figure 6. A decommissioning management organization.

7.2. Approaches to decommissioning

The initial requirement of decommissioning is to determine the right approach for decommissioning. The three different strategies normally found in the literature are:

- (a) Storage with surveillance
- (b) Restricted site use
- (c) Unrestricted site use

The final decision concerning which strategy, or combination of strategies, will be followed will be made at the time of decommissioning. However, it is felt that, keeping in mind the partial decommissioning experience, option **c)** will be easily achievable, at least to the extent that the reactor building can be used unrestricted.

7.3. Better record keeping

The main purpose of the records maintained during the operation of PARR-1 has been to satisfy the requirements of the national nuclear regulatory authorities. However, now PARR-1 records also include the information (based on the experience from partial decommissioning and from IAEA CRP), which may prove helpful in decommissioning. The details of the relevant records are as follows:

- the operating history of PARR-1, including any changes in the core geometry;
- spills or inadvertent releases of radioactive material/contamination that might affect decommissioning;
- modifications to the facility;
- operating and maintenance records of systems and equipment;
- the design and location of experimental devices used during the lifetime of PARR-1;
- radiation survey data;
- system and facility drawings;
- photographs;
- process and operating manuals; and
- special activities and techniques developed during the partial decommissioning of the reactor.

It has been learned from the experience of partial decommissioning that due emphasis must be given to archiving the system drawings, facility drawings, and documentation of special activities and techniques developed during the partial decommissioning of the reactor. Furthermore, a record classification system, based on the mandatory minimum time to keep that record, was introduced. Consequently some of the records have been properly archived and will be kept for the life of the facility.

After such a long time, the design documents have started showing signs of decay/wear. To cope with this problem the design documents were redrawn with AutoCAD[®]. The others were scanned and stored on computer-readable compact disks (CDs). The existing paper copies are reproduced and stored for the intended life of the facility. The special procedures adopted for removal of each reactor component during the partial decommissioning were documented. Photographs and videos of such activities were also made to facilitate the training of manpower and future decommissioning of PARR-1.

7.4. Rehiring of trained staff

Recognizing the importance of plant knowledge and experience of the staff, particularly those involved in the installation and/or partial decommissioning of PARR-1, it was decided to rehire some of the retired staff on an as-needed basis. To attain this goal it is decided to keep track of the retired personnel, who have worked on key assignments on PARR-1, particularly during partial decommissioning.

7.5. Interim spent fuel storage bay capacity

As mentioned previously, an interim spent fuel storage bay has been developed. This bay has been designed to store 400 irradiated fuel and reflected elements.

After the upgrade of the reactor power, the increase in flux has decreased the irradiation time of samples by a large amount because a central gap has been provided for these irradiations. The flux in the central gap is of the order of 10^{14} n/cm²/sec. Figures 7–9 show the operation of the reactor during the last four years. These figures show that the time of operation has reduced over the years. The length of irradiations has been reduced due to the increase of flux, better management and the modernization of the radioisotope production facility i.e. the procedures of radioisotope production have been changed from wet to dry.

With the present frequency of spent fuel introduction into the interim spent fuel bay (approximately 3 spent fuel elements per year) the total spent fuel elements, including the last core (about 30 elements) will be about 90 to 120 fuel elements in the next 20 to 30 years respectively. Including the HEU spent fuel elements (84), already stored in the interim spent fuel bay, these become equal to about 170 to 200 fuel elements. Calculations are presently being performed to analyze the possibility for the reuse of HEU fuel elements previously not fully utilized. The operation of the reactor core with mixed fuel elements would further decrease the number of fuel elements to be introduced in the storage bay.

The final storage of the fuel elements has not been considered yet. However, work on a fuel repository has been started.

7.6. Enhancement of waste disposal facilities

It is felt that liquid waste from the reactor can be dealt with easily, as was done during partial decommissioning of the reactor (Figure 10), and hopefully will not pose any problem. However, the solid waste facilities need to be enhanced.

Previously the solid waste was disposed off in 4 m deep trenches of different sizes. The practice was to cover the trench with clay when it is filled to two third of its depth. The area was fenced and was approachable only by the radioactive waste management personnel.

Recently, about two years ago, some funds were made available for enhancement of the waste facilities. Consequently a compactor was purchased to compact the solid material by the use of an hydraulic press in standard 200 litre MS drums (Figure 11). An engineered trench was constructed measuring approximately 15×3×4 m (Figure 12). The trench, when filled up to 3 m, is capped with clay and concrete. In addition to this, a small sized cementing facility was also made available allow the cementation of radioactive waste drums.

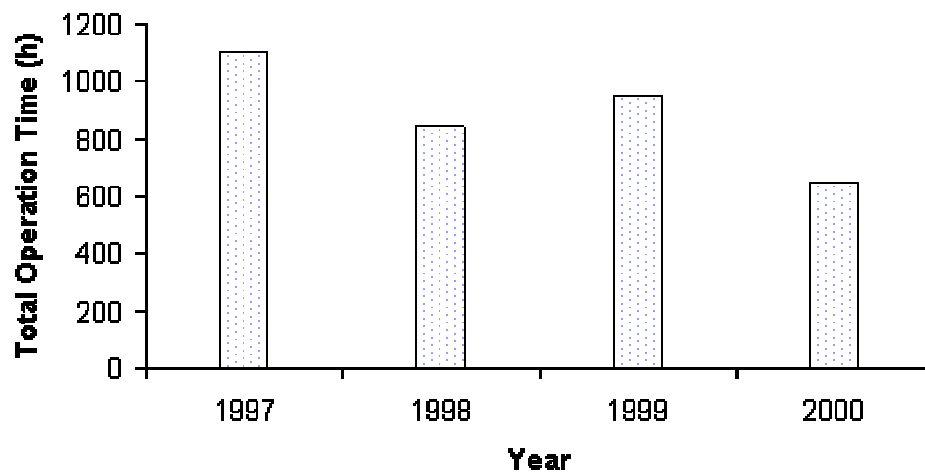


Figure 7. PARR-1 total operation time (history).

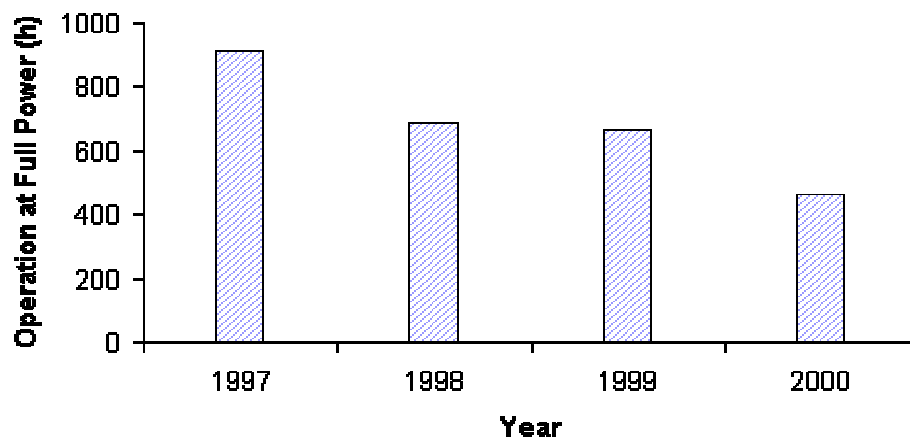


Figure 8. PARR-1 full power time (history).

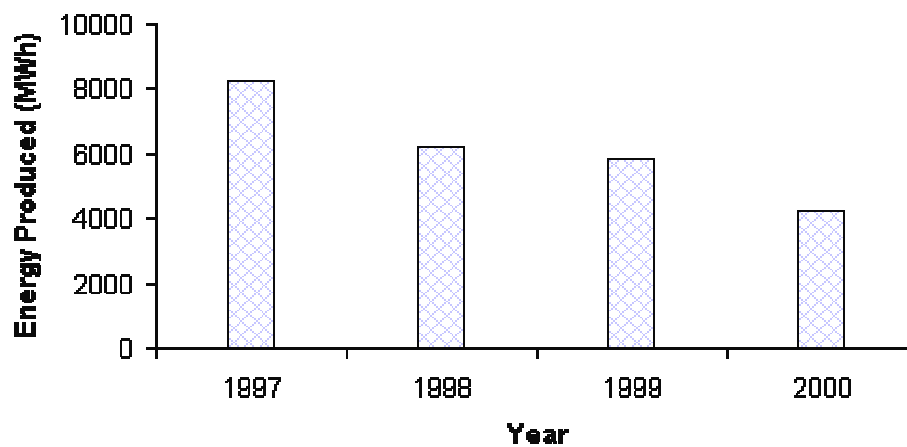


Figure 9. Energy produced during PARR-1 operation (history).



Figure 10. Tanks for storage (for decay) of low-level liquid radioactive waste.

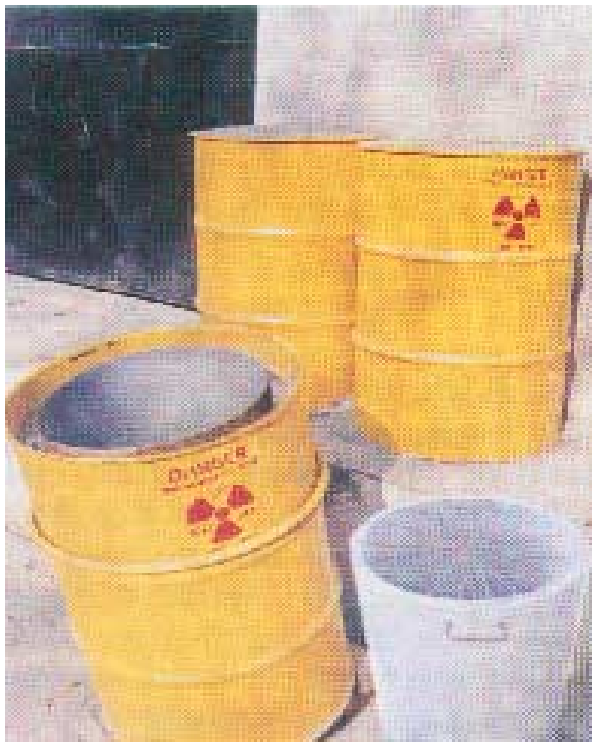


Figure 11. Multi-barrier MS and concrete drums.

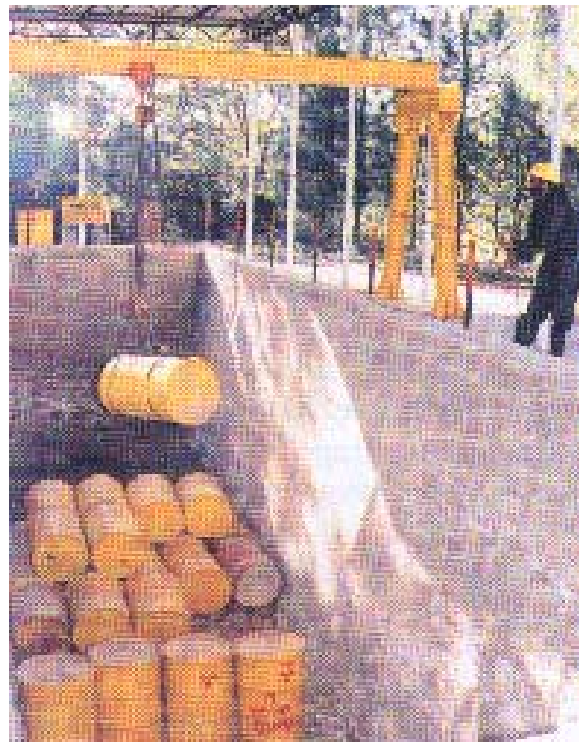


Figure 12. Trench for solid waste.

A proposal has been submitted to the relevant authorities for development of radioactive waste technologies at the Pakistan Institute of Nuclear Science and Technology where PARR-1 is located. The technologies planned for development are:

- separation of radionuclides of importance by chemical methods;
- evaporation;
- compaction/super compaction;
- cementation;
- bitumization;
- vitrification; and
- containment for spent fuel elements.

7.7. National repository for disposal of nuclear waste

A proposal for the design, development and construction of a national repository for disposal of packages of radioactive waste has been initiated. A 10-year plan has been proposed to the concerned authorities.

8. Conclusions

The partial decommissioning and restart of PARR-1 provided an experience to work with the reactor fuel and components having high radiation fields. It was also an experience to handle large volumes of solid and liquid radioactive waste. The whole operation was completed without any overexposure or contamination of the workers due to sound planning, careful monitoring and supervision. No uncontrolled releases of radioactivity to the environment took place during the whole operation. Though the work was carried out under some financial constraints and non-availability of sophisticated tools, safety of workers and environment was ensured by the use of remote handling tools, shielding and pre-planned exposure reduction procedures and adherence to the ALARA principle. Within the country these practices of radioactive waste management have received approval of the National Regulatory Body and safety experts. The upgraded reactor has been licensed by the National Regulatory Body to be operated at 10 MW(th). The experience has demonstrated that safety objectives and criteria can be met. This experience will prove useful in even better planning and execution of future decommissioning of PARR-1 in particular, and for other Pakistani research reactors like PARR-2 (27 kW MNSR), in general.

The worldwide activities on decommissioning, particularly those communicated through the IAEA CRP, have brought the awareness of some aspects/actions to be implemented regarding planning for decommissioning at an early stage. In this regard, it is worth mentioning that the importance of early steps, i.e., record keeping, rehiring of trained staff, and interim and final waste storage has been well recognized. Such steps have already been initiated at PARR-1.

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Appendix A

DECOMMISSION PROCEDURES FOR BEAM TUBES

- A.1. AUTHORIZATION** : Head ROG
- A.2. RESPONSIBILITY** : Reactor Supervisor
- A.3. MANPOWER** : Three reactor operators, two technicians, one health physicist and two laboratory attendants.
- A.4. INSTRUMENTS, TOOLS, MATERIAL REQUIRED** : Tools for removal and installation of plug collimator, nuts, and flanges. One stainless steel blind flange and screw jack for the removal of beam tube.
- A.5. HAZARDS** : (a) Radiation field will exist in the working area;
(b) Possibility of industrial hazards exists.

A.6. <u>GENERAL PROCEDURES</u>	<u>REMARKS/INITIAL</u>
1. Fill proformas ROG-22* and HP-2*	_____
2. Arrange tools, spanner set, socket set, screw jack, polyethylene sheet etc.	_____
3. Arrange S.S. blind flange	_____
4. Move the bridge to the open end.	_____
beam-tube lev	
7. Drain beam tube-1. Remove lead and concrete plug from tube.	_____
8. Proforma radiation monitoring.	_____
9. Remove the nuts of beam tube flange.	_____
10. Apply screw jack and remove the beam tube.	_____
11. Measure maximum dose rate and wrap the tube in polyethylene sheet.	_____
12. Store beam tube at N-5 hole in reactor hall area.	_____
13. Fix the blind flange and tighten the nuts.	_____
14. Fill the stall pool slowly and check for seepage/leakage around the blind flange.	_____
15. Insert lead plugs and fix the end cover.	_____
16. Remove pool-dividing gate.	_____
17. Move the bridge to stall end.	_____

* Particular Proforma which are to be filled for any activity in the reactor hall for approval of Head Reactor Operation Group & Head Health Physics Division.

Principal results of “DEWAM” project implementation, Russian Federation

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Abstract. This overview summarizes the key results of the four-year project “Decontamination and Waste Management in the Course of Research Reactors Decommissioning” carried out within the framework of the IAEA’s CRP “Decommissioning Techniques for Research Reactors”. The project included two principal components: (1) info-analytical studies and development of a database system, and (2) research and development in the areas of decontamination and waste management technologies applicable to decommissioning. Details of the work are expounded in 29 publications and annual Progress Reports; the results of the study are used in corresponding university courses; and innovative technologies for radwaste processing and environmental restoration are planned to be introduced into practice.

1. Introduction

The main objective of this contribution to the IAEA Co-ordinated Research Project (CRP) on “Decommissioning Techniques for Research Reactors” was to create a decommissioning database in terms of operating experience, technologies available in the Russian Federation (RF), etc. This CRP coincided with a national programme being undertaken in Russia, the “Decontamination and Waste Management in the Course of Research Reactors Decommissioning” or DEWAM project and was factored into the content of the project, as discussed below. As a result, not only is the database discussed but also the background on much of the data it contains. Examples of Russian experience in decommissioning research reactors are discussed along with development projects for several different decommissioning and waste conditioning methods with practical examples of their use.

Russia has prolonged, multiform and rather extensive practical experience in decommissioning of research reactors and other nuclear facilities, the first projects being in the earlier fifties. However, right up to the late eighties, state technical policy on decommissioning had not considered such important issues as the strategy, regulation, techniques, funding, etc., and each D&D project was dealt with as a first of a kind project.

There were several purposeful co-ordinated activities in the period from 1986 to 1991 (provoked, to some extent, by the Chernobyl accident), which were suspended by the geopolitical and socio-economic reforms in the former USSR. Developments in this sphere were not revived until the middle of the nineties.

Thus, at the time of the launching of the IAEA Co-ordinated Research Project (CRP) on “Decommissioning Techniques for Research Reactors” many aspects of the national decommissioning programme (Figure 1) were and still are in the active stage of evolution — development, selection/optimization, concordance and official approval.

Understanding that regulatory, economic and administrative decisions may dictate technical requirements for decommissioning, and in turn – existing technological bases may influence the choice of preferable strategic approaches, it would be unreasonable to consider techniques out of context with the general situation.

This objective circumstance predetermined the content and the main trend of the research project “Decontamination and Waste Management in the Course of Research Reactors Decommissioning” (DEWAM) carried out at the St. Petersburg State Institute of Technology (SPIT), Engineering Radioecology and Radiochemical Technology Department.

In particular, the scope of the project included two major missions carried out simultaneously and in the close interdependence:

First, info-analytical investigations to (a) monitor the overall situation of “transferred period” with feasible participation in the decision making process through advice to authoritative agencies, expert appraisals, improvements of educational and training programmes, etc., and (b) provide the necessary tools and information support for comparative engineering analyses through the creation of a corresponding database system;

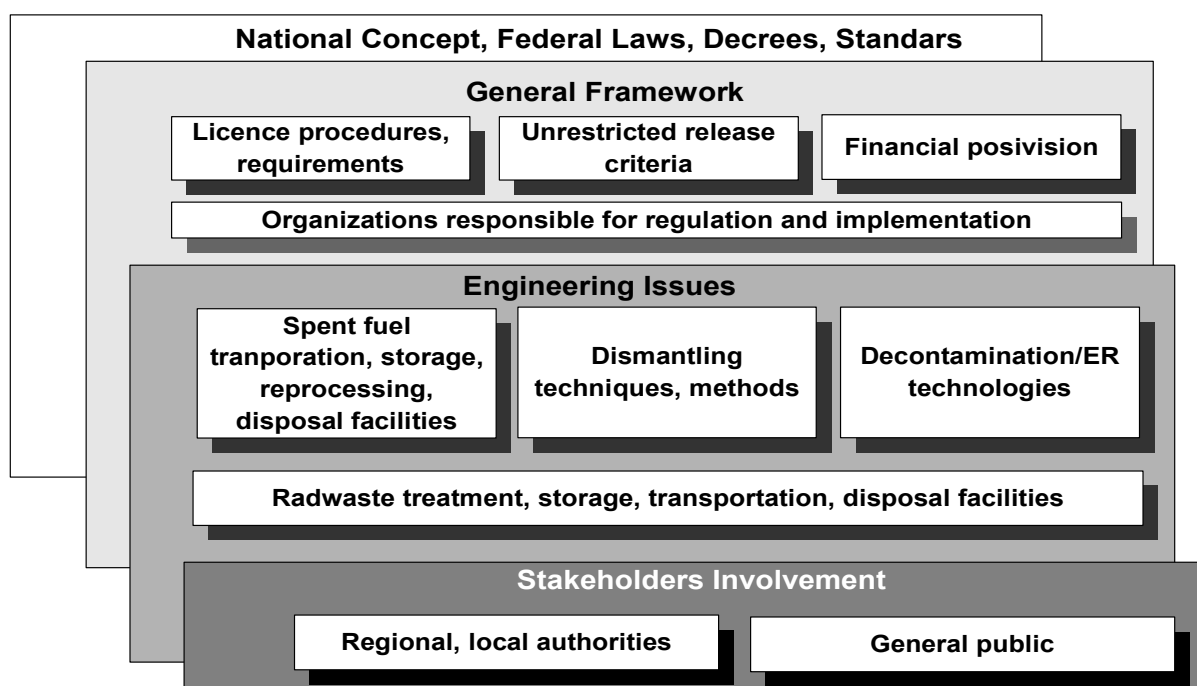


Figure 1. Principal components of decommissioning programme.

Second, research and development (R&D) in the areas of decontamination and management of the radioactive waste already accumulated in the facilities intended to be decommissioned or that expected to be generated during the decommissioning process.

Such an approach enabled the project to address several key areas of decommissioning important for the harmonization of the national D&D programme, and yet still conform to the terms of reference for the IAEA’s CRP. It is important that, in terms of the present project (focused on decontamination and radwaste management), the decommissioning of research reactors (RR) is considered as a part of a global program, rather than as a specific, independent area of activity. In other words, in general, programs developed and tested for RR, must be applicable to other nuclear installations.

The principal results of the CRP related studies are expounded in 29 publications, and included in the SPIT courses of lectures on “Radioactive Waste Management” and

“Decontamination of Nuclear Facilities and Radioactively Contaminated Sites”. A reference book on “Decommissioning of Nuclear and Radiation Facilities” is planned to be completed and issued in early 2002.

This paper represents a consistent overview of the recent developments and achievements in the field rather than a detailed description of motivations, procedures of works, and results of the separate subprojects implemented within the framework of the CRP. Because of the space limitations, this paper cites only those publications prepared by the CRP participants. The full list of references includes more than 200 entries.

This work was supported by the IAEA (302-T2-RUS-9681.3), Minatom RF (1.02.28.2000.055), and RF Ministry of Education under Grants TOO-7.4-2766 and TOO-7.6-2763.

2. Computerized D&D info-analytical system

In the absence of a centralized data base on research reactors and more or less comprehensive catalogues on domestic decontamination technologies and radwaste management techniques consideration was given to the fact that info-analytical support of D&D activities could offer several important advantages to:

- (1) **planning bodies** — clarity of the level of technical readiness for implementation of D&D programs; understanding of the uncertainties to be encountered, and long term priorities of technical policy; improved basis for realistic planning in the financial sphere; a real base for objective competitive selection of technologies and techniques to be introduced into practice;
- (2) **regulatory authorities** — understanding of, and opportunities to react timely to possible needs/implications/deficiencies/imperfections in legislation and regulatory control;
- (3) **licensees** — opportunities for proper planning, selection of optimal technical solutions, the quest of convenient partners, etc.;
- (4) **technology developers** — clarity of the actual directions for new developments and/or necessary improvements to be competitive in the D&D market.

Under these circumstances the aim of subproject was defined as follows:

- to collect and systematize information on research reactors, decontamination technologies, radioactive waste management techniques, and materials recycling methods (both market tested and innovative) applicable for decommissioning purposes;
- to develop a system of criteria and procedures for comparative analysis of the data collected;
- to create a computerized multicriteria decision – making tool which will allow determination of the appropriateness of various techniques (or combinations of techniques) to perform D&D programs.

The research reactors database contains rather detailed information about 42 nuclear installations including the reactor’s name, type, startup date, power, neutron flux, location, and design features (fuel, reactor, active core, control and protection systems, equipment, biological shielding). In addition, the results of statistical processing of the data compiled are presented for comparative analysis.

The decontamination and radioactive waste management database system consists of several main databases: “Inventories” (facilities, operational waste, fuel, materials), “Technologies” (general information, cost data, operational data and implementation data) and “Closing” (storage, disposal and recycling) as well as auxiliary databases “Directory of Companies”, “Geo-information”, “Documentary Support” and “Bibliography”.

The “Decontamination technologies” files describe more than 100 commercially available or innovative techniques including information on areas of application, the essence of the methods, conditions for implementation, technical characteristics, decontamination factor, producer/designer and comments. These files are supplemented with drawings where possible.

The “Radwaste management” files include data on methods, areas of application and necessary technical details. A few tens of radwaste processing and conditioning technologies potentially applicable for D&D purposes (including environmental restoration methods) are compiled and analyzed.

Auxiliary databases include: "Directory of Russian Organizations Involved in Environmental Protection Activities", comprising 750 institutions; “Regulation”; “Disposal Opportunities”; and the specialized files “Bibliography of Publications” and “Dismantling Techniques”.

A Database Management System (Figures 2, 3) allows one, in principle, to manipulate and analyze this data for the selection, optimization and practical application of proper decontamination technologies and radioactive waste management techniques complying with up to date regulatory requirements, and storage/disposal capacities. Currently the systematized information is used for educational purposes, and the first reference book on decommissioning is planned to be prepared on this basis.

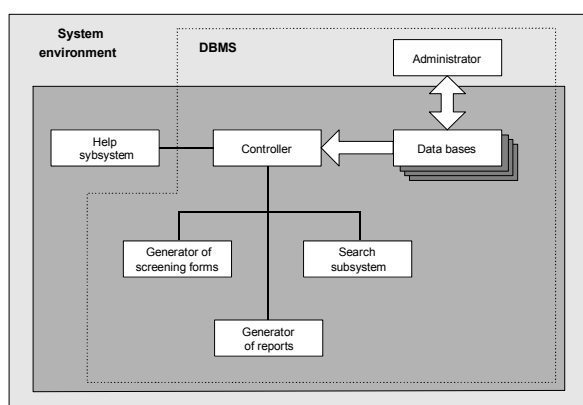


Figure 2. Principal structure of software support: content and interaction of modules.

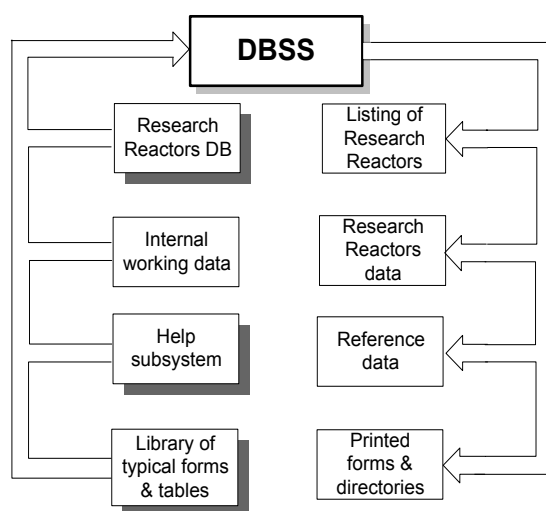


Figure 3. Information routes scheme (database software system). An example.

Moreover the methodologies, procedures and criteria developed for comparative engineering analyses were acknowledged by the state NPPs operation agency, ROSENERGOATOM, as an effective tool for the optimization of nuclear power plant radwaste processing/conditioning technologies, and included in the state programme.

3. Decommissioning of research reactors: Practical experience

This section (a part of the info-analytical activities) is devoted to a summary of decommissioning projects at research reactors (RRs). It is aimed, first of all, at demonstrating the diversity of RRs construction features and the variety of decommissioning options available. Not the least of the motives is also the aspiration to dispel a widespread delusion that decommissioning in Russia is, allegedly, an undeveloped nuclear sector. And, finally, that practice is the only criterion for the determination of the viability of theoretical views.

Unfortunately it was not always possible to obtain verified information on some of the important details of the decommissioning projects studied. However, general approaches, techniques used, radiation situation and other data compiled provide analysts with quite complete information for the objective assessments of the situations and for reasonable planning for the actions required.

3.1. Experimental radioisotopes production reactor IR

The first practical experience in decommissioning-related activities was in the middle fifties when the experimental uranium-graphite reactor, IR, was partly dismantled and inspected after 4 years of operation.

3.1.1. Design features of the IR reactor

Design features of the IR reactor include:

- fuel – 2% enriched uranium;
- material of moderator and reflector – graphite blocks;
- active core and reflector – vertical cylinder constructed from graphite blocks;
- roofing – aluminum alloy blocks on the 0.5 mm layer of cadmium;
- the graphite block stack was tied up by 10 arrays of steel bands and enclosed in a steel housing;
- integral neutron flux in the center of the core – $4.5 \cdot 10^{21} \text{ cm}^{-2}$.

3.1.2. Dismantling

The technology for dismantling was developed by a special group formed from operational personnel. About 40 unique tools and devices were designed and manufactured for this purpose.

Specifically, temporary shielding for dismantling operations was constructed from 10 mm thick sheet steel in the form of a cylindrical tank 3.8 m diameter and 2.14 m high. The tank had a through passage, the opening of which could be covered in part by movable vessels filled with water. The tank was installed over the facility, in place of the upper shielding, and flooded with water up to the 1.5 m level, in such a way that it was possible to accomplish dismantling operations through the exposed part of the opening. The device used for temporary storage of the dismantled upper shielding also resembled a metal tank 2.8 m high with double walls. The thickness of the protective layer of water, in this case, was 900 mm. For the removal of graphite blocks from the stack, bars with unclenched claws were designed along with special conductors to clamp the surrounding blocks.

Dismantling operations commenced with the decontamination of the upper structures and high-pressure washing of the coolant systems and lower structures. The cooling gas through the graphite was stopped. Then the uranium and radioisotopes were unloaded from the core; following by the removal of the control and protective rods, experimental tubes and the upper shielding, and the installation of the water-filled tank as a temporary shield. After that a special conductor was placed in the vicinity of the graphite column intended to be removed. Adjacent columns were fastened to the conductor with bars. To prevent the spilling of radioactive graphite dust and pieces on the roofing, a metal tray with a funnel was installed on the conductor. A cover was put on the funnel for the collection of the graphite pieces.

The graphite blocks were removed from the core column by column. To achieve this a bar was brought into a channel of the graphite stack; the claw was let down in a metal bushing, and the bushing, together with the column, was removed from the core. If dosimetry data indicated abnormally high activity on the column, the graphite blocks were removed one by one and placed immediately into a protective shielding container. All together 40 columns containing 400 graphite bricks were removed, including 21 columns from the damaged areas of the graphite stack.

3.1.3. Dosimetry and radiation protection

Before the start of dismantling operations, a special programme of dosimetry measurements was developed, and the necessary equipment, additional means of individual protection and dosimetry devices for general and individual dose control were obtained.

Detectors were placed at various points in the core, including the space between the upper shielding and roofing of the facility; between the graphite blocks and the lateral water shielding; under the graphite and over the drain tank. As a rule, two detectors from two different dosimetry facilities were placed at each point of measurement.

The upper structures of the facility had practically no induced radioactivity. Surface contamination did not exceed $74\text{--}1000 \text{ mBq} \cdot \text{cm}^{-2}$ ($2 \times 10^{-12} - 3 \times 10^{-11} \text{ Ci} \cdot \text{cm}^{-2}$) generally. In the area of the damaged cells the specific surface activity reached $7.4 \text{ kBq} \cdot \text{cm}^{-2}$ ($2 \times 10^{-7} \text{ Ci} \cdot \text{cm}^{-2}$). The dose rate at a distance of 0.5 m from the bottom of upper shielding was around $1 \mu\text{Sv} \cdot \text{s}^{-1}$ ($100 \mu\text{R} \cdot \text{s}^{-1}$) and increased to $2 \mu\text{Sv} \cdot \text{s}^{-1}$ ($200 \mu\text{R} \cdot \text{s}^{-1}$) at locations corresponding to the damaged cells. The dose rate from the upper shielding, placed in the storage tank after dismantling, did not exceed $10\text{--}20 \text{ nSv} \cdot \text{s}^{-1}$ ($1\text{--}2 \mu\text{R} \cdot \text{s}^{-1}$). Installation of temporary (operational) shielding over the open core decreased the dose rate from $2 \mu\text{Sv} \cdot \text{s}^{-1}$ to $0.25 \mu\text{Sv} \cdot \text{s}^{-1}$ ($200 \mu\text{R} \cdot \text{s}^{-1}$ to $25 \mu\text{R} \cdot \text{s}^{-1}$), with a dose rate at the assembly opening of $0.5 \mu\text{Sv} \cdot \text{s}^{-1}$ ($50 \mu\text{R} \cdot \text{s}^{-1}$). The strongest sources of γ radiation were the damaged graphite bricks: the dose rate at a distance of 0.5 m from a damaged block came to $600 \mu\text{Sv} \cdot \text{s}^{-1}$ ($60000 \mu\text{R} \cdot \text{s}^{-1}$) while it did not exceed $0.80 \mu\text{Sv} \cdot \text{s}^{-1}$ ($80 \mu\text{R} \cdot \text{s}^{-1}$) in the undamaged area of the brickwork from which the block was removed.

A high level of radioactive aerosol contamination of the premises accompanied dismantling operations. During the removal of the damaged graphite blocks specific radioactivity in the air reached $7.4 \text{ kBq} \cdot \text{dm}^{-3}$ ($2 \times 10^{-7} \text{ Ci} \cdot \text{dm}^{-3}$), and the intensity of precipitation accounted for $8\text{--}500 \text{ kBq}/(\text{m}^2 \cdot \text{h})$ ($2.2 \times 10^{-7} - 14 \times 10^{-7} \text{ Ci}/(\text{m}^2 \cdot \text{h})$). Plastic coated clothes and respirators with goggles were used to protect personnel from the radioactive dust and aerosols. When necessary personnel were provided with breathing air masks connected to a supply of clean air. Protective overalls were replaced several times per shift.

When leaving the working hall, employees were subject to a preliminary decontamination and contamination checkpoint in a washing station situated at the exit from the hall. Then, outside the protective gate to the working hall, employees were again treated in a special extension. Here they removed the protective overalls, and then passed through a stationary contamination checkpoint.

Strict dose control was implemented at all work sites and in the working hall as a whole. The floor in the hall, covered with stainless steel sheet, was washed repeatedly until all the surface contamination was removed. A large number of pointers and explanatory pamphlets were used to inform the workers. All the employees were provided with individual electrodosimeters and individual film badges. The maximum permissible level of exposure was set at 0.5 mSv (50 mR) per shift. On average, the radiation exposure of personnel during the 10-month period of the work was below the permissible limit.

3.2. Heavy water research reactor TVR

The heavy water research reactor, TVR, at the Institute of Theoretical and Experimental Physics was commissioned in 1949 and operated for 37 years — until 1986. The design power of the reactor was 500 kW; after reconstruction it was increased to 2.5 MW. In 1986 the reactor was inspected, and then shutdown for decommissioning.

From 1988 to 1997 the following decommissioning-related actions were undertaken:

- (1) Radiation surveys were carried out for the core, support systems and facilities for temporary storage of radioactive waste. The composition and specific activities of reactor materials, and the surface contamination of equipment, pipelines and experimental devices were determined.

Documents entitled “Technical and Economic Substantiation of TVR Reactor Decommissioning” and “Technological Process of Reactor Structures Dismantling” were developed

On the basis of the results of the radiation survey and the documents prepared the following decommissioning operations were carried out: (1) unloading of the spent nuclear fuel from the core to the water storage pool with subsequent transportation to the radiochemical complex “Mayak”; (2) an inventory of heavy water; (3) dismantling of equipment outside the concrete shielding; (4) decontamination of the reactor site.

In the course of decommissioning an important problem encountered was the tritium generated in the heavy water during the operation of the reactor. The specific activity of the heavy water, drained from the reactor and the primary cooling system, was 0.23 TBq/L. Inner surfaces of the reactor tank and all the piping connected to the tank were subjected to vacuum drying.

A special facility for purification of heavy water from the mechanical admixtures with the aid of fibreglass filters was installed in the support systems compartment of the reactor. After purification, the heavy water was placed in standard 30 L stainless steel drums. At present these drums are stored in the warehouse of the Institute (without processing of the heavy water). Because of the long lifetime of tritium ($T_{1/2} = 12.3$ year), and the high permeability of tritium vaporous, the problem of heavy water management in the course of TVR decommissioning continues to be topical.

After unloading the spent fuel the following equipment was dismantled: (1) pipelines and components of the primary cooling system; (2) pipelines and components of the gas

cooling system for the graphite; (3) vacuum systems of the reactor tank, primary and gas circuits; (4) control systems, and experimental and auxiliary equipment.

Both standard and specially developed tools and facilities were employed in the dismantling operations. All the work was carried out in isolated enclosures equipped with exhaust-fans. Additionally, the workstations for gas or mechanical cutting of metal were equipped with mobile ventilation facilities connected to the main exhaust-fan system.

After removal of the spent reactor fuel from the 6 m deep storage pool, 55 tons of radioactive light water remained in the pool. This water was treated and transported to SIA “Radon”. The pool was decontaminated using specially developed technology and remote equipment.

One of the most radiation-intensive operations was the removal of horizontal channels, made in the form of “pockets” (300 mm length; 200 mm diameter), from the surface of the inner vessel of the reactor. As the dose rate in the area of these channels was 250 $\mu\text{Sv/h}$, a special machine was developed and manufactured which carried out these operations. This machine was installed in the reactor on a securely fixed base at a depth of 4 m. The machine was equipped with a crown milling cutter and could be moved remotely in the horizontal plane.

During reactor operation, (particularly, in the early years, when the centralized radwaste storage facility was absent and experience in radioactive waste management was rather limited), some radioactively contaminated areas ($P_d=30 \mu\text{SV/h}$) were created on the reactor site. Therefore, together with the dismantling activity, environmental remediation operations have been carried out since 1990. As a result more than 200 tons of radioactive soil and asphalt-concrete waste have been removed and transported to the disposal site of SIA “Radon”.

In general, the decommissioning project has been carried out successfully without radiological impact on the environment and population, with a limited number of employees involved, and with full adherence to the national standards and rules of radiation safety.

3.3. Heterogeneous research reactor WWR-2

The WWR-2 reactor is a heterogeneous, thermal neutron facility. Water is used as the coolant, moderator and upper shielding of the reactor. The core loading of the WWR-2 is 4.5 kg of ^{235}U ; the maximum neutron flux in the center of active core is $4 \times 10^{13} \text{ n}\cdot\text{cm}^{-2}\cdot\text{s}^{-1}$. The active core of the reactor (a cylinder 400 mm in diameter and 500 mm high) consists of assemblies of 10 mm diameter fuel rods. Structural components of the reactor (piping, grids, tank, etc.) are made from aluminum alloys. The initial power of the reactor, as commissioned in 1954, was 300 kW; after reconstruction in 1957 it's power was increased to 3 MW.

In 1983 the WWR-2 reactor was shutdown for reconstruction aimed at meeting new safety requirements and targets for new programmes of experimental research. However, under public pressure it was decided to start decommissioning operations instead of reconstruction. During its 29 years in service, it had operated for about 150,000 hours.

To provide the necessary infrastructure for dismantling activities, an on-site storage facility for unloaded spent fuel and high-activity structural components of the reactor was constructed in 1983. In addition: (1) the building for the contamination checkpoint for personnel was reconstructed; (2) the post for radiation and dosimetry control was upgraded

with the necessary equipment; (3) special tools for dismantling operations as well as handling and transportation devices were obtained; (4) special compartments for temporary storage and conditioning of radwaste generated in the course of decommissioning were organized.

The dismantling operations were carried out in the following sequence: (1) investigation of the radiation levels to be encountered during the dismantling with the subsequent estimation of radiation doses; (2) unloading and transportation of spent nuclear fuel to the special on-site storage facility; (3) decontamination of the primary circuit of the reactor; (4) consecutive dismantling of equipment, pipelines and metal structures in the reactor hall and pumping module; (5) demolition of the concrete shielding and removal of contaminated soil with the sorting and transportation of waste to the temporary storage facility or to specially prepared areas at the reactor site; (6) removal of non-radioactive waste.

After unloading the fuel and decontaminating the primary circuit, the following systems were dismantled: reactor tank, rolling basket, structures of the water-shielding tank and cast-iron blocks of the biological shield. Since the level of radioactivity of the equipment and shielding had decreased considerably in the ten years since the reactor was shutdown (1983), standard methods with some shielding adjustments were used in the dismantling operations. To be specific, the metal structures were cut up using oxy-acetylene, plasma and contact-arc cutting, and the handling and moving of the pieces was carried out with the aid of slings, cables, pulleys, hand-winches and crane. The dismantling of the metal exhaust-pipe (42 m high and 0.85 m diameter) was carried out in two stages: four upper sections of 30 m total length were brought down, and then these and the lower part on the foundation were cut into transportable pieces with an oxy-acetylene torch. These cylindrical pieces had bottoms attached and were then used as containers for the radioactive waste removed from the reactor site.

The total weight of the dismantled equipment, pipelines and metal structures (including the components of radiation shielding) came to around 630 tons, of which 600 tons was contaminated or had an induced activity and had to be treated as radioactive material.

The maximum level of gamma radiation from the cast-iron elements of the biological shielding, located near the active core, was about 3 Sv/h. Operational liquid radioactive waste (total volume – 14 m³; total activity – 2.6×10^{10} Bq) was stored on-site in metal tanks. Solid radwaste (total mass – 70 tons; total activity – 3.1×10^{12} Bq) is in storage in the special building. The rest of the solid radwastes were transported to Moscow SIA “Radon” for processing and disposal. The spent nuclear fuel is currently located in the water pool of the on-site storage facility, and will eventually be sent to “Mayak” for reprocessing.

During the five months of D&D operations the radiation doses that personnel were exposed to did not exceed the official limit of 50 mSv/a. Continuous dose measurements around the perimeter of the reactor site indicated that there was no radiation impact on the surrounding population from the decommissioning work.

3.4. Reactor physical technical RPT

The uranium-graphite RPT reactor in Moscow was commissioned in 1952 with an initial power level of 10 MW. After reconstruction in 1957 its power level was increased to 20 MW. In 1962 it was decided to decommission the RPT reactor, and to construct a new research reactor MR (materials testing reactor) in the same building.

After unloading the nuclear fuel and dismantling the support systems the frame of the RPT reactor together with the graphite stack was sealed “in situ” in concrete.

The solid radioactive wastes (primarily the elements of support systems) were placed in deep concrete modules in the reactor building. Low level liquid radwastes were stored, without processing, in metal storage tanks.

In 1963 the multi-loops reactor MR was placed in operation in the same building. This experience – “in situ” disposal of a reactor after fuel unloading and dismantling of equipment with subsequent construction of a new reactor facility on the same place – was also used in 1993 as a decommissioning option for the SM-2 reactor.

3.5. Research reactor RG-1M

In 1970 an analytical complex to be used for neutron activation analysis of raw materials and the end products was placed in operation at the Norilsk mining and metallurgical complex. The complex included a 100 kW research reactor; five radiochemical and two radiometric labs. A pool-type reactor, RG-1M, was placed in a 1.6 m diameter by 4.8 m deep concrete pit lined with stainless steel. The reactor pool (1.5 m diameter, 3.5 m high) manufactured from titanium alloy and filled with demineralized water contained an active core (60 fuel pins and 30 graphite reflectors), the channels for regulation and protection systems, experimental and thermometric channels, ion chambers and some other auxiliary equipment. The primary cooling system included a heat exchanger, a pump and filters for coolant purification. The reactor was covered with a 430 mm thick cast-iron plate, equipped with rotary plugs. The neutron flux in the active core was $2.3 \times 10^{12} \text{ n} \cdot \text{cm}^{-2} \cdot \text{s}^{-1}$.

In 1998 it a decision was made to decommission the reactor complex.

3.5.1. Radiation situation

Based on the results of experimental investigations and radiological computations the accumulated activities include: $5 \times 10^{11} \text{ Bq}$ — for 2200 kg of reactor components; $1.3 \times 10^{11} \text{ Bq}$ — for the reactor pool liner; $7.4 \times 10^{11} \text{ Bq}$ — for the concrete; and $3.7 \times 10^{10} \text{ Bq}$ — for all equipment in the radiochemical laboratories. Activity of the coolant did not exceed $10^3 - 10^4 \text{ Bq} \cdot \text{dm}^{-3}$; surface β -contamination of the primary cooling system was $100 - 200 \text{ cm}^{-2} \cdot \text{min}^{-1}$. During construction of the facilities for the collection and storage of liquid radwaste the dose rates varied from 4 to $18 \mu\text{Sv} \cdot \text{h}^{-1}$ (over the drainage pit); the walls were β -contaminated up to $20 \text{ cm}^{-2} \cdot \text{min}^{-1}$; the flow — $150 \dots 300 \text{ cm}^{-2} \cdot \text{min}^{-1}$; external surfaces of the waste storage containers — $100 \dots 250 \text{ cm}^{-2} \cdot \text{min}^{-1}$.

3.5.2. Decommissioning programme

The decommissioning concept and programme have been developed on the basis of comprehensive engineering and radiation surveys.

The first stage of the DECOM programme — defueling the reactor and shipment of the spent fuel to the radiochemical plant “Mayak” — was carefully planned and successfully carried out in 1999. The subsequent stages of decommissioning include:

- dismantling and conditioning/containerization of in-reactor equipment and structures for transportation to the radwaste storage/disposal enterprise “Radon”;
- alteration of the existing radwaste storage facility (RWSF) and the reactor shaft into stationary repositories for on-site disposal of both solid and liquid radioactive waste.

RWSF is an underground reinforced concrete compartment about $6 \times 9 \text{ m}$ in area and 5.5 m deep with a 5 mm thick steel liner and one upper inlet. This facility (as a part of the

reactor complex) is situated in a permafrost mountain-mass, far from the dwellings and industrial buildings of Norilsk. In the RWSF six installed LRW storage containers occupy about 10% of the space.

Operational and decommissioning low level radwaste (a total mass of about 5000 kg) containerized in 200 L metal drums will be placed in the compartment such that the distance between any container and the nearest wall is greater than 0,5 m. The total number of containers in a compartment will be around 50. After the drums are in place they will be surrounded with concrete up to their tops and covered with a concrete “pillow” 300 mm thickness. The upper layer of the “pillow” will be hydroisolated. After that 1.5–2.0 m of the bedrock will be spread over the repository to prevent the thawing out of the soil in the summer period. For radiological monitoring observation boreholes 10 m deep are arranged around the RWSF. It is envisaged that institutional control will also be applied.

The second facility for radwaste disposal is the shaft with the reactor tank inside from which all the equipment has been removed and the coolant drained. The shaft is also concrete, and covered by a 500 mm concrete “pillow” with hydroisolation and by 1.5–2.0 m of the bedrock.

Safety analysis demonstrates that radioactivity will not penetrate the engineering barriers for at least 300 years, and, in general, the disposal facilities meet present regulatory requirements. All decommissioning activities are carried out under the control and supervision of local and central offices of the state nuclear inspection agency.

3.6. Training reactor facilities VM-A and VM-4

Two prototype PWR nuclear power facilities (NPF) with power levels of 70 and 90 MW thermal, commissioned in 1968 and 1983 respectively, were operated at the USSR Navy’s Training Center (Paldisky, former Estonian Soviet Republic) until 1983 when both reactors were shutdown for decommissioning. Each NPF, located in a special reactor compartment, included the reactor itself, steam generators, pumps and other systems of the primary and auxiliary circuits. All necessary equipment for process control was placed in adjacent compartments. Both NPF were located in a common stand hall, 180 m long, 18 m wide and 22 m high, equipped with two electrical bridge–cranes of 50 t hoisting capacities.

3.6.1. Radiation situation

The NPFs were operated in accordance with the regular programme of training courses, a regime of operation containing rather frequent shutdowns. Normally the operating power level did not exceed 20–30% of the nominal ones. There were no incidents or even any deviations from the planned operating regime. There were no radioactive contamination incidents recorded of the reactor sites or adjoining areas during their operation.

The dose rates encountered were:

- 0.12 and 0.15 $\mu\text{Sv}\cdot\text{h}^{-1}$ — in accessible areas of VM-A and VM-4, respectively;
- 1.1 and 0.2 $\mu\text{Sv}\cdot\text{h}^{-1}$ — in the reactor rooms;
- 1.9 and 0.23 $\mu\text{Sv}\cdot\text{h}^{-1}$ — on top of the reactor.

The total accumulated radioactivity (without nuclear fuel) was assessed as 1×10^{15} Bq in VM-A and 3.9×10^{14} Bq in VM-4 with the following percentage of radionuclide composition: ^{60}Co — 31, ^{55}Fe — 58, ^{59}Ni — 1.2, ^{63}Ni — 10.3. 99% of this activity was concentrated in the reactor itself and in the iron water-shielding tank.

This information was confirmed by the results of an independent radiation survey carried out by American specialists in 1995.

3.6.2. Decommissioning programme

As the first inevitable stage of decommissioning, the nuclear fuel was removed from the Training Center and sent to the radiochemical plant “Mayak” for storage and possible reprocessing.

An International Expert Group (formed by the initiative of Estonian Government) with participation of specialists from Russia, IAEA, and a number of Western European countries critically analyzed a multivariate concept of the All Russian Institute of Complex Power Technology, and selected a safestore option as the most preferable temporary decision for the next 50 years.

Conservation of the reactor compartments (RC) was carried out as follows:

- (1) the RCs were separated from the adjacent compartments which, in turn, were completely dismantled;
- (2) After unloading the fuel standard lids were put on the reactors and welded to support frames; all the openings in the lids were hermetically sealed as well;
- (3) the reactors and the primary cooling systems were drained; sorbents from the filters of the coolant purification system were removed; all pipes related to the primary circuit and drainage systems were sealed; some components and structures, located above the biological shielding, were dismantled;
- (4) reinforced concrete shelters were constructed around both RCs. In addition, for shielding purposes, around 30 m³ of concrete were poured in RC N1, and 41 m³ in RC N2; the air inside the RCs was dried, and external surfaces of RCs were painted with a special corrosion-resistant enamel.

The NPFs were thus transformed into a safe storage state, provided with three radiation protection barriers: (1) hermetically sealed primary cooling systems, (2) hermetically sealed frame and bulkheads of the reactor compartments, and (3) construction of shelters capable of withstanding such external impacts as an earthquake of magnitude seven, an air percussion wave, or the fall of heavy objects, etc.

The RCs do not require maintenance, active control or an energy supply for the duration of the storage period. Periodic radiation measurements inside the shelter and air sampling can be done through special penetrations in the walls of the shelters.

After five-years of observations the storage conditions are characterized as "normal".

Since 1999 Russian experts have participated in the development of a programme on the RCs management after completion of the temporary storage period. Three main options are under consideration: (1) on-site disposal of the RCs without dismantling; (2) dismantling of RCs with subsequent conditioning and disposal of the radioactive waste generated; (3) prolongation of the “safestore” period for up to 300 years.

The final solution has not yet been decided.

4. Lessons learned

Experience gained in research reactor decommissioning (or decommissioning related activities) covers practically all the options potentially available and officially approved, including:

- partial dismantling, inspection, decontamination and following reconstruction intended for *facility life extension* (see 3.1);
- *care and maintenance regime* accepted at some research reactors in expectation of dismantling operations;
- partial dismantling with *in situ disposal of the reactor* and subsequent construction of an advanced apparatus in the same reactor building (see 3.4);
- dismantling of reactor and equipment, demolition of buildings with *on-site temporary storage of radioactive waste* (see 3.3);
- complete dismantling of facility with environmental remediation of the reactor site – *green field state* (see 3.2);
- decontamination and dismantling of reactor and radiochemical laboratories with *on-site disposal of radioactive waste* (see 3.5);
- preparation of reactors for “passive” long term storage — “*safestore*” concept (see 3.6).

The choice of a D&D option depends on the cost of the project, which may vary widely (Table I). However economics is not the most crucial reason. As usual the *combination of technical, social (including public pressure) and political factors dictate the final solution*.

A number of lessons can be learned about reactor materials behaviour from practical decommissioning experience. This information is important for proper organization and implementation of dismantling operations. As examples one could mention the following:

Table I. Comparative costs assessment of decommissioning options*

	Expenditures	Decommissioning after safestore period (years)				
		5		30	50	100
		without decont.	decont.	decont.	decont.	decont.
1	Decontamination of premises, equipment and pipings: decontamination systems operational expenditures	- -	1.0 0.5	1.0 0.5	1.0 0.5	1.0 0.5
2	Deep decontamination of scrap metals: techniques operational expenditures	0.35 0.80	0.35 0.80	0.35 0.80	0.35 0.80	0.35 0.80
3	Preparation for safestore: capital investments running expenditures	0.90 2.24	0.90 2.24	2.88 3.20	3.88 4.40	0.91 5.60
4	Maintenance of facility	-	-	8.74	15.73	17.00
5	Supply of remote equipment	30.0	30.0	16.0	15.0	14.0
6	Dismantling: reactor equipment, piping, etc.	31.0 9.81	31.0 3.27	22.3 2.97	18.50 2.70	16.50 2.50
7	LRW processing	0.03	0.31	0.31	0.31	0.31
8	IRW processing	4.18	4.18	2.09	1.46	1.05
9	Radwaste disposal	9.48	9.48	7.11	6.64	5.69
10	Total	89.23	84.03	68.25	71.30	76.27

* Costs are given in arbitrary units.

- under the “normal” operation of uranium-graphite RRs the areas where the maximum changes to graphite properties occur is on the periphery of the core near the upper and lower bounds of the stack;
- the uranium from a damaged fuel element remains in the clearances between the blocks and between the columns within an area encompassing a few cells around the damage point. The greatest concentration of uranium is observed in between the ends of the blocks. Because of that it is impossible to completely remove the uranium. During reactor operation this uranium accelerates degradation of the graphite and contributes to radioactive contamination of the cooling gas;
- serious corrosion damage takes place at the points of the contact between aluminum and stainless steel, etc.

Both domestic and foreign decommissioning experience clearly demonstrates that “*as-built*” drawings do not always reflect current conditions and sometimes do not include important data, e.g. information on the weight of components that may have to be handled during decommissioning, or on the chemical composition of materials, etc. To overcome possible problems:

- it is advisable to start a decommissioning project as soon as possible after shutdown while the memories of operating personnel are fresh enough to validate support documentation;
- the drawings have to be checked as much as possible, e.g. by visual inspection;
- personnel need to be continuously reminded that when they find conditions different from those expected, the work must be stopped and management consulted before work can continue;
- sufficient time must be allocated for the development of the decommissioning strategy and preparation of a decommissioning plan;
- decommissioning tools should be designed and fabricated with sufficient flexibility, etc.

The most important concerns, in the projects discussed, relate to decontamination and radioactive waste management issues, which may give rise to serious difficulties in D&D activities caused by *the lack of appropriate methods or techniques and the lack of capacity for storage and/or disposal of decommissioning waste*. The overall results clearly demonstrated the validity of such concerns.

5. Advanced processing technologies

In accordance with national regulations, obtaining a license for decommissioning (or even for facility life extension) requires one to demonstrate that all the radwastes accumulated at a reactor site can be removed or transferred to an environmentally safe form, and that all waste expected to be generated during the decommissioning process will be adequately managed.

Typical liquid radioactive wastes are a rather complex composition of non-radioactive substances (toxic or harmless, chemically active or inert) containing a very small weight percentage of radionuclides. This circumstance has to be taken into account because the presence of large amounts of “secondary” compounds in the radwaste may lead to:

- (i) the needless increasing of the volumes of conditioned waste by including non-radioactive macrocomponents in the final form;
- (ii) the limitations in selection of appropriate isolating matrices owing to the specific properties of chemical compounds; and
- (iii) the serious difficulties in extraction/isolation of radionuclides.

Such a situation is typical for “historical” wastes from research reactors, which normally have a complicated character in respect to chemical composition, specific activities and concentrations of organics (Table II).

The obvious objectives of processing technologies for these wastes are:

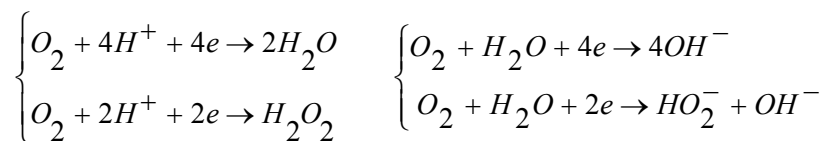
- (i) the separation of macrocomponents (solids and water) in a form which allows them to be stored without special precautions (preferably as non-radioactive waste) or to reuse them in technological processes; and
- (ii) the concentration of radionuclides in the smallest volume possible and in a form which reliably isolates the activity.

The presence of complexing organics makes it practically impossible to employ such traditional techniques as adsorption or ion-exchange for the isolation of polyvalent radionuclides. Careful analysis of the data available and a critical review of the fundamental physico-chemical interactions indicates that organics may have a strong influence on the effectiveness of liquid radwaste decontamination from monovalent ^{137}Cs , as well. Generation of neutral associates of cesium with organic anions is initiated by sodium nitrate, and makes ^{137}Cs inactive in respect to selective sorption. The amount of such associates is assessed to be of about 0.1% of the total cesium concentration, but it means that the removal of organics from the radwaste may dramatically increase the decontamination factor (by a few orders of magnitude). Thus removal of complexing organic constituents is one of the key preconditions of the treatment of liquid radwaste.

Table II. Inventory of liquid evaporator concentrates stored at AM reactor

Parameter	Content				
	Tank OV-175		Tank OV-176		
	1991	1996	1991	1996	1997
Density, $\text{kg}\cdot\text{m}^{-3}$	-	1.28	-	1.23	1.24
pH	>10	13.6	>10	13.3	-
Dry remainder, $\text{g}\cdot\text{dm}^{-3}$	550	540	376	400	410
Na, $\text{g}\cdot\text{dm}^{-3}$	-	130	-	110	90
K, $\text{g}\cdot\text{dm}^{-3}$	-	3.2	-	2.3	3.0
Fe, $\text{g}\cdot\text{dm}^{-3}$	1.55	-	1.35	-	-
CCO , $\text{g}(\text{O}_2)\cdot\text{dm}^{-3}$	252	156	155	118	100
NO_3^- , $\text{g}\cdot\text{dm}^{-3}$	196	177	142	150	195
Cl^- , $\text{g}\cdot\text{dm}^{-3}$	14.4	18.0	6.3	14.0	11.4
SO_4^{2-} , $\text{g}\cdot\text{dm}^{-3}$	-	27	-	20	-
Silicoacid, $\text{g}\cdot\text{dm}^{-3}$	29.3	-	12.8	-	-
Soaps, $\text{g}\cdot\text{dm}^{-3}$	16.5	13.0	10.0	8.0	2.5
Oils, $\text{g}\cdot\text{dm}^{-3}$	4.5	-	12.5	-	-
Extracted greases, $\text{g}\cdot\text{dm}^{-3}$	3.1	32	5.7	14	33
Anionic surfactants, $\text{g}\cdot\text{dm}^{-3}$	162	122	117	75	41
^{137}Cs , $\text{Bq}\cdot\text{dm}^{-3}$	$4.7\cdot 10^8$	$3.8\cdot 10^8$	$4.2\cdot 10^8$	$3.8\cdot 10^8$	$4.8\cdot 10^8$
^{90}Sr , $\text{Bq}\cdot\text{dm}^{-3}$	-	$<10^3$	-	$<10^3$	-
α -emitters, $\text{Bq}\cdot\text{dm}^{-3}$	$3.8\cdot 10^4$	$5.3\cdot 10^4$	$2.0\cdot 10^4$	$1.9\cdot 10^4$	-
Uranium, $\text{Bq}\cdot\text{dm}^{-3}$	$2.0\cdot 10^3$	$6.3\cdot 10^3$	$<10^3$	$1.3\cdot 10^3$	-
Plutonium, $\text{Bq}\cdot\text{dm}^{-3}$	$2.4\cdot 10^3$	$5.7\cdot 10^3$	$<10^3$	$1.3\cdot 10^3$	-
Americium, $\text{Bq}\cdot\text{dm}^{-3}$	$3.2\cdot 10^4$	$4.1\cdot 10^4$	$7.7\cdot 10^3$	$7.7\cdot 10^3$	-
Curium, $\text{Bq}\cdot\text{dm}^{-3}$	$1.6\cdot 10^3$	$<10^3$	$1.2\cdot 10^4$	$8.4\cdot 10^3$	-

The method of electro-stimulated destruction (ESD) of organics has been proposed to achieve this end. The original idea of the ESD method is based on the following reactions for cathodic reduction of molecular oxygen:



The anode of the ESD cell is fabricated from lead dioxide. The gas-diffusive cathode is made from a special hydrophobic electro-conductive material able to absorb of about 500 volumes of oxygen; the oxygen coming from air pumped into the gas chamber of the system. The principal advantage of ESD is the intrinsic safety of such a “destructor” as compared to water electrolysis with the generation of hydrogen.

The effectiveness of the ESD method is compatible with traditional methods (ozonation, electro-chemical oxidation). Its advantages are:

- energy consumption is reduced by a factor between 6 and 20 (depending on the nature of compounds destructed);
- the equipment needed is about 40 times smaller;
- initial investments and operational expenses are lower;
- it can be applied to solutions containing up to at least 400 g·dm⁻³ of salts content; and
- it guarantees that no “ratting mixture” is generated.

Pilot-scale trials have clearly demonstrated that, in some cases, the employment of the ESD method for processing of radioactive liquid concentrates removes the necessity for any ion-exchange decontamination stage, since polyvalent radionuclides are separated:

- (a) by co-precipitation and subsequent filtration, and
- (b) by selective sorption together with cesium.

The volume of the concentrates treated can be reduced 200–500 times and most of the activity can be transformed into a solid form. The residual activities of ⁶⁰Co and ¹³⁷Cs in the product water are less than 3 Bq·dm⁻³ (below detection limit of the analytical equipment used) while initial concentrations were 4.0 × 10⁵ and 6.3 × 10⁶ Bq·dm⁻³, respectively. Electricity consumption is about 20–25 (A·hour)·dm⁻³. This scheme was successfully employed for the treatment of borate solutions containing the evaporator residues from the NPP, with the WWER type reactor, and now an “advanced version” of the scheme is being tested for application to the “historical” waste from research reactors.

The “advanced version” is a self-supported, self-regulated ESD system that does not require any external energy source. Cells with gas-diffusive cathodes and special anode-adsorbents have been constructed and investigated in depth under laboratory conditions. Unlike the cells with an external power supply, these systems have somewhat lower productivity and consume pure oxygen. Nevertheless, an independent, self-regulated destructor provides obvious advantages for rendering both radioactive and chemically toxic wastes. At present the “hot” tests are being conducted at the AM reactor in Obninsk.

The destruction technique, developed in SPIT, offer several advantages that are inherent in the system. First, the oxidation/destruction processes are accomplished at near ambient pressure and temperature. Second, all waste stream components and most oxidation products

are contained in an aqueous environment that acts as an accumulator for any inorganics which were present in the original waste stream, and also provides a thermal buffer for the energy released during oxidation of the organics. Third, the generation of secondary waste is minimal, as the process needs no additional chemicals. Finally, the entire process can be shut down by simply turning off the power (or shutting off the oxygen supply), affording a level of control unavailable in some other techniques.

However, the wide diversity of radwaste (with regard to chemical and radionuclides composition; properties and concentrations of inorganics contained; presence of insoluble deposits, complexing organics, etc.) does not permit the effective use of the same technology for all the waste streams. Thus it is logical to have a set of functional modules that enable one to choose the appropriate technological approach, depending on the waste compositions and the ultimate goals of processing.

To develop this concept the various experimental capabilities, theoretical ideas, practical experiences and creative energies of the investigators from SPIT, the Alexandrov Research Institute of Technology and St. Petersburg State University have been collected within the framework of the St. Petersburg R&D Initiative (SPRDI). The intentions of the SPRDI Group are to develop an industrial mobile facility equipped with a family of flexible functional modules. These plans are based on the results of preliminary R&D that has already provided essential information needed to resolve the problem.

In particular, the technology and modular unit for treatment (MUT) of low level radwaste has been developed and tested in semi-industrial conditions. MUT involves basic functional modules for microfiltration, ultrafiltration, reverse osmosis and ion-exchange, as well as auxiliary modules for the radwaste preparation, softening, and spent resin regeneration. The modules are arranged in two purification lines (Figure 4).

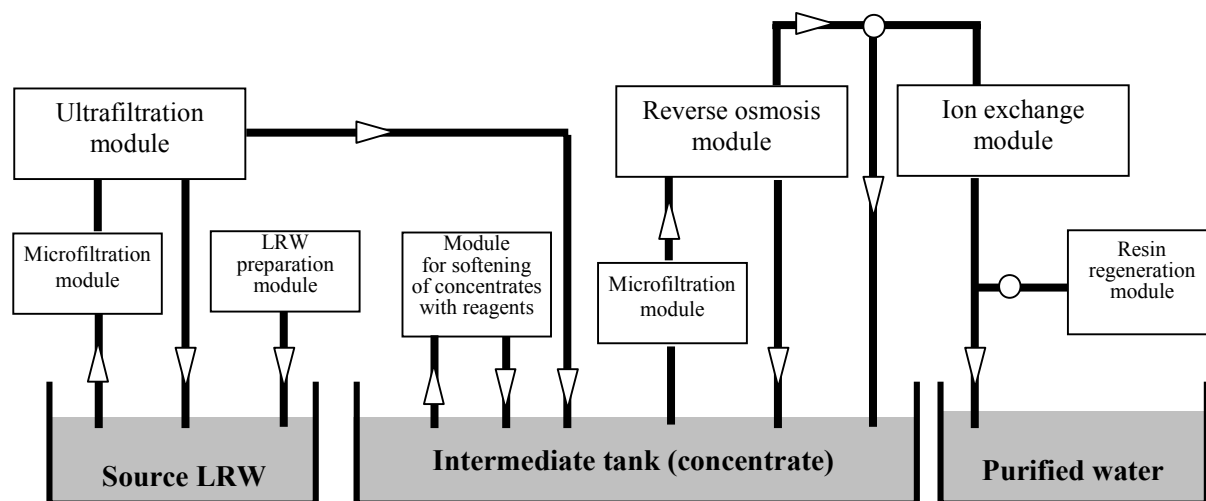


Figure 4. Flowchart of the modular membrane-sorption unit for concentration of liquid radioactive waste.

The capacity of MUT is 0.5 m³ per hour. The product water is subjected to continuous radiation and chemical monitoring. MUT has treated more than 500 m³ of liquid radioactive waste with the following characteristics: 10 — 100 mg·dm⁻³ of suspensions; 1 — 43 mg·dm⁻³ of corrosion products; 1.3 — 27 mg·dm⁻³ of ammonia; 1 — 24 mg·dm⁻³ of oil products; 0.24 — 0.36 mg·dm⁻³ of surfactants. The initial specific activity was between 3.7·10² and

$3.7 \times 10^4 \text{ Bq} \cdot \text{dm}^{-3}$; the final product had less than $37 \text{ Bq} \cdot \text{dm}^{-3}$ (the National Standard maximum permissible level for ^{90}Sr).

In addition to MUT, the facility described above includes a modular unit for conditioning (MUC) which converts the radioactive concentrates into a stable solid form and packages it in suitable waste containers (Figure 5)

The unit provides for the cementation of compounds directly in barrels, which are fitted with built-in mixers. The process does not require the handling of cement slurry and minimizes the risk of radioactive contamination in the work areas. The concentrates that arise in MUT are solidified with a solution/cement ratio of about 0.7, and with the addition of 10 weight % of environmentally innocuous sorbents (clay, vermiculite, zeolite, etc.).

The maximum release of radiocesium into the environment (e.g. in the case of accidental flooding of the storage/disposal facility) is assessed to be no more than 2% of the initial activity for Portland cement; 1% for slag Portland cement, and 0.3% for alumina cement (leaching rate comes to $1 \times 10^{-5} \text{ g} \cdot \text{cm}^{-2} \cdot \text{day}^{-1}$). The leachability of heavy metals is 50 to 100 times lower than that of cesium. This solidified waste meets the regulations for storage in engineered facilities and for disposal in shallow repositories.

The prototype modular facility has already been successfully employed for LRW processing at several enterprises in the Far East and the Northeast regions of Russia. To expand its usage and to improve treatment technology and/or safety and reliability of the conditioned radwaste, a number of other options are under consideration within the framework of SPRDI.

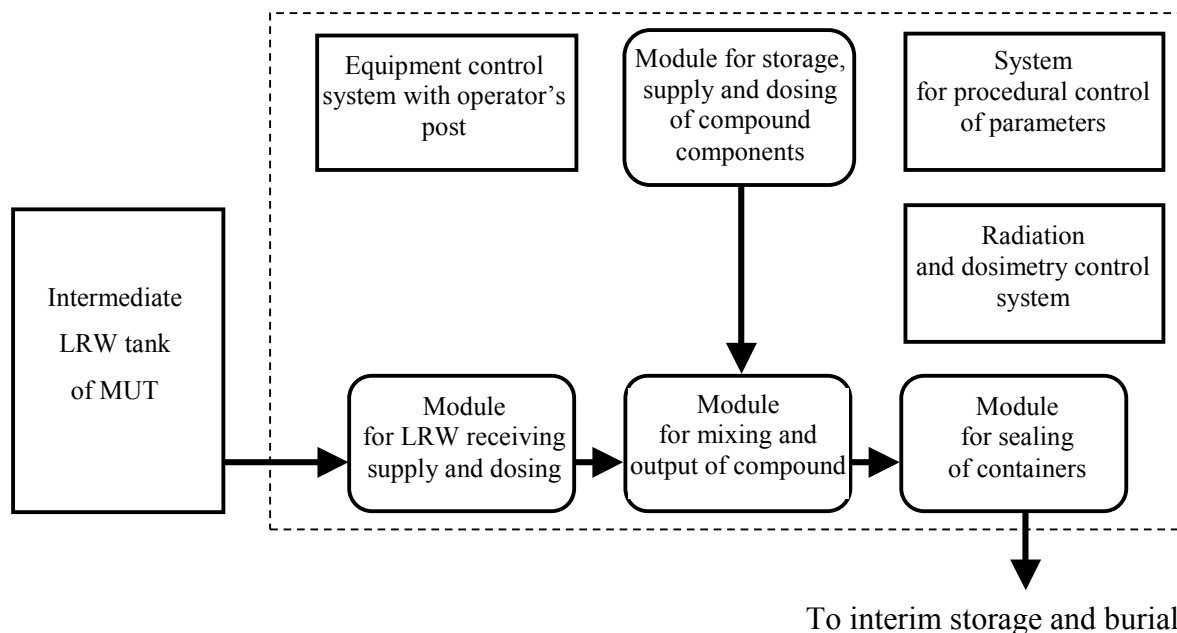


Figure 5. Flowchart of the LRW solidification unit.

An epoxy-acrylic composition was selected as a prospective material for solidification of radioactive concentrates. This composition can be produced in air by ionizing radiation at ambient temperature with technologically acceptable doses. After conversion to a 3D-form

epoxy-based compositions acquire unique properties: high thermal resistance (up to 540 K); good physico-mechanical characteristics (compression strength up to 240 MPa, bending strength up to 1200 MPa; Brinell hardness up to 250 MPa); excellent water and oil resistance; high radiation stability (at least up to 100 MGy); and low leachability (less than 10^{-6} g·cm⁻²·day⁻¹). The thermal conductivity of the composition is between 0.5 and 1.0 W·m⁻¹·K⁻¹, and can be increased by an order of magnitude by the addition of boron nitride.

In order to decrease the dose required for final “solidification”, i.e. formation of a stable 3D-structure, special stabilizers based on 3-ethylen-4-amin (TETA) can be used. Experiments have demonstrated that the dose for final solidification can be decreased to 1.0–1.5 kGy with the addition of 1–3% TETA, while the initial TETA-doped epoxy-acrylic composition is viable (suitable for application) for 30 days after mixing.

It is important to emphasize that solidification does not required an external radiation source. The process is carried out by exposure to the ionizing radiation of the radwaste incorporated in it. Stable forms can include up to 60% of radioactive concentrates.

The simplicity of this technology and the good performance characteristics of the conditioned radwaste have stimulated efforts to develop corresponding equipment for subsequent use as “solidification” modules either in the existing MUC or as an independent facility.

A considerable volume of data has been obtained on the incorporation of simulated radioactive waste (typical metal oxides with addition of ~1.7% of Sr, and 5.35% of Cs) into silicate ceramic matrices additionally protected with glass-like or ceramic-type isolating coatings.

The matrix itself was a mixture of calcine with quartz and Al₂O₃ in proportion 1.0:0.84:0.16. A protective coating was prepared from naturally occurring materials typical to the Northwest region: granite and nepheline syenites incorporated in the tails of apatite ore with sodium carbonate being added as a flux. After cold pressing the samples were sintered at 900 °C for one hour and cooled down to room temperature in an air-hardening mode.

Research indicated that protective coatings of the standard 10–12 mm thickness synthesized from the naturally occurring materials could ensure safe contact of the ceramic radwaste forms with underground waters for several hundred years, i.e. for a period sufficient for the decay of ⁹⁰Sr and ¹³⁷Cs. Adaptation of the laboratory methods to industrial conditions (preferably to the conditions of a mobile facility) is considered as one of the objectives of SPRDI.

An engineering analysis of innovative technologies already carried out within the framework of SPRDI confirms the validity and practicality of the approach adopted. Variations of the number, succession and predestination of technological modules enables one to provide optimal conditions for the treatment and conditioning of practically any waste in accordance with the set goals which, in turn, can be different depending on the waste type, prehistory of the waste storage, transportation and disposal requirements.

In addition, in the framework of the DEWAM project, rather promising results have been obtained on managing such specific decommissioning waste as activated graphite that had been contaminated with fission products. This technology involves covering the decontaminated graphite blocks with composite metallic coatings. Potentially, “conditioned” graphite waste can be stored in simplified facilities or disposed of in geological formations.

However additional “hot” tests are required to demonstrate any advantage of this method over others.

The CRP project’s group has also contributed to the development of industrial technology for deep treatment of metal waste created during the decommissioning of nuclear installations. A new facility for decontamination remelting of metal scrap was put into operation in April 2001.

6. Decontamination technology for environmental restoration of nuclear installation sites

At present there exists a broad spectrum of technologies for decontamination of both man-made and natural objects. The database created within the framework of the DEWAM project includes more than 100 domestic methods and techniques potentially applicable for decommissioning purposes but not always commercially available and often requiring modification to be employed in a given decommissioning project.

Under these circumstances the efforts of the CRP project’s group in the decontamination area were logically concentrated on the solution of concrete problems by means of reasonable adaptation of known technologies or combinations of methods and techniques (with necessary improvements and/or modifications) to the site-specific conditions.

Two such projects deserve mention in the context of the present overview. These are:

- (1) the design, development, and fabrication of a facility for decontamination of surface and drainage water collected at the radwaste storage and disposal site of SIA “Radon”; and the development and “hot” testing of soil-washing technology for environmental remediation of a decommissioned laboratory site.

It is clear that both situations (contamination of soil and drainage water), in one way or another, could be typical of research reactor sites.

The facility for the purification of surface and drainage waters contaminated with ^{137}Cs , ^{90}Sr and plutonium involves an electrochemical cell with a dividing electroconductive membrane; a cascade of mechanical filters (quartz filter and polymer filter with a pore size from 1–20 μm); and a cascade of sorption filters based on the strong cation exchanger KU-2-8 and the selective sorbent “Ferrocyanide NZA”. It was demonstrated that the expensive synthetic sorbents NZA with a relatively short working cycle can be successfully replaced by modified natural sorbents based on mordenite, clinoptilolite or glauconite.

Semi-industrial trials reliably demonstrated the effectiveness of the technology developed. In particular, decontamination factors were defined as 150–500 for ^{137}Cs ($A_{\text{in}} \leq 5.6 \times 10^3 \text{ Bq}\cdot\text{dm}^{-3}$); 100–150 for ^{90}Sr ($A_{\text{in}} \leq 1 \times 10^3 \text{ Bq}\cdot\text{dm}^{-3}$) and more than 300 for plutonium ($A_{\text{in}} \leq 2 \times 10^3 \text{ Bq}\cdot\text{dm}^{-3}$). Capacities of natural Cs-selective sorbents and ion-exchangers were assessed to be about 10^3 and $(5-6) \times 10^2$ column volumes, correspondingly.

The second project, the decontamination of soil, has been carried out for environmental restoration of the Shkipersky Protoc site situated almost in the downtown of St. Petersburg at Vasilievsky Island. As a result of past bad practice, the urban soil (sand, clay, builder’s refuse, peat and the scraps of wood) is contaminated with ^{137}Cs and ^{90}Sr up to a level of $1 \times 10^5 \text{ Bq}\cdot\text{kg}^{-1}$. The distribution of activities is close to equilibrium with some concentration

of radionuclides in the organic constituents. Leachability of cesium and strontium by the natural water is nondetectable on a real time scale.

Soil-washing technology was proposed as an alternative to soil excavation with subsequent transportation and disposal of more than 10000 ton of waste at Leningrad Special Enterprise “Radon”.

In laboratory experiments the following factors, which influence the effectiveness of decontamination, have been investigated and optimized: composition of washing solutions, temperature of mixing, the rate of passing the solution through the column with soil, ratio of solution’s volume to the mass of soil, and concentration of reagents.

As a result, 4–5% aqueous solution of HCl was selected as an optimal washing fluid with the soil/solution ratio of 1.3:1. The spent decontamination solution was recycled after electrochemical regeneration. Secondary wastes were neutralized and evaporated to dryness.

Application of this technology decreased the mass of radioactive waste from 10000 tons to approximately 50 tons (secondary decontamination waste and some soil). Less than 3000 tons of soil must be transported to the “normal” dust heap, and the rest is suitable for unrestricted use.

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Decommissioning strategy for reactor AM, Russian Federation

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Abstract. This paper presents the results of studies into the various aspects of decommissioning the oldest Russian research reactor, the AM reactor. Experimental and calculation results of a study to determine the inventory of long lived radioactive materials at the AM reactor are presented, along with a comparison to comparable data for other similar reactors. An analysis, by calculation, of the decay time needed to allow manual dismantling of the reactor vessel and stack, without remote operated equipment, defined it as 90 years. The possibility of burning most of the irradiated graphite to decrease the amount of long lived radioactive wastes was confirmed. The problems associated with the dismantling of the reactor components, contaminated with radioactive corrosion products, were analyzed. A decommissioning strategy for reactor AM was formed which is deferred dismantling, placing most of the radiological areas into long term safe enclosure. An overall decommissioning plan for reactor AM is given.

1. Introduction

The AM is a uranium-graphite research reactor with water coolant, sited in Obninsk (near Moscow). The reactor was started up in June 1954 and operated with an average power of 6.4 MWt for 47 years. The final shutdown of the reactor is planned for December of 2004. Table I presents the reactor operating power level history for the AM reactor, which has been used for the investigation of new types of fuel assemblies, and the production of radioisotopes and radiopharmaceuticals. A cut-away view of the reactor is shown in Figure 1. The graphite stack, 3 m in diameter and 4.5 m high, is the main structural element. Its central part, 1.5 m in diameter and 1.7 m high contains the core. The core is comprised of 151 six-sided graphite columns with 65 mm diameter central holes which contain 128 fuel assemblies and 23 control rods. The cells of the graphite stack form a triangular lattice with a pitch of 120 mm. The thickness of the graphite reflector is 75 cm in the radial direction, and 70 cm at the top and bottom. A radiation shield over the top of the reflector consists of a 140 cm graphite layer and 70 cm of cast iron. The cast iron radiation shield over the coolant pipes is 20 cm thick. The reactor stack is contained in a steel vessel surrounded by a radial radiation shield consisting of 100 cm of water in a tank and 300 cm of concrete. An annular reinforced concrete shield, 45 cm thick by 120 cm high, surrounds the top of the reactor vessel in the water tank. The steel bottom plate of the reactor vessel is supported by a cast iron and concrete cooling base sitting on the bottom concrete body. The weight of radioactive reactor materials is given in Table II.

2. Project objectives for AM

The objectives of the programme on the problems of decommissioning the AM reactor, performed as part of the IAEA CRP on Decommissioning Techniques for Research Reactors during 1998–2001, were:

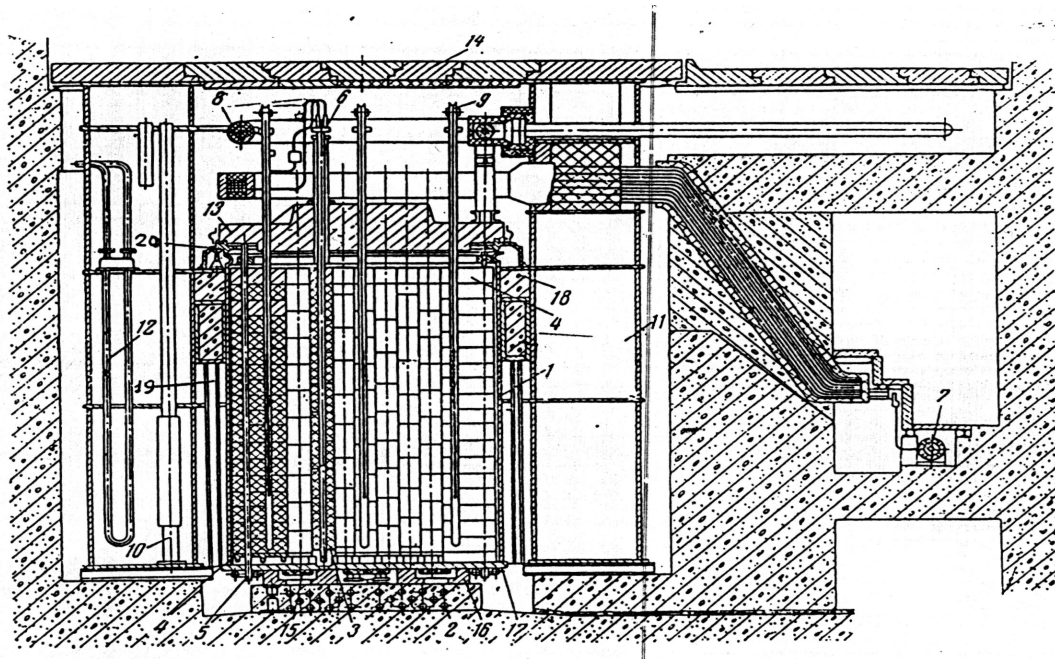
- (1) estimating the inventory of long lived radioactivity in the defueled reactor by performing radiometric measurements of samples of structural materials, chemical measurements of microimpurities contained in them, and corresponding physical calculations;
- (2) decay calculations;

Table I. Reactor AM operating history

Period	Time, years	Mean power, Mwt(th)	Comments
1954–1971	16.5	7.4	
1971	0.75	0	Repair work
1971–1986	15.25	6.6	
1987	1.0	0	Updating
1988–2001	12	6	
2001–2004	5	6	Planning

Table II. Radioactive materials in AM reactor

Material	Weight, t
Graphite	50.8
Cast iron	165.8
Steels	87.2
Concrete	66.6
Lead	8.5



1. Reactor vessel 2. Concrete base cooling 3. Core 4. Neutron reflector 5. Neutron reflector cooling 6. Fuel assembly 7. Inlet header 8. Outlet header 9. Control rod 10. Ionization chamber 11. Water shield 12. Water shield cooling 13. Top plate 14. Top cast iron shield 15. Bottom base cooling 16. Bottom base and it's support 17. Bottom plate 18. Top concrete shield 19. Support of top concrete shield 20. Vessel flange.

Figure. 1. AM reactor.

- (3) calculation of estimates of the dose rates to be incurred during the dismantling of the reactor vessel and graphite stack ;
- (4) calculation of estimates of the dose rates to be incurred during the dismantling of the reactor vessel and graphite stack, without using remotely operated manipulators, in order to define the required decay time for deferred removal;
- (5) experimental studies of decontamination methods for the primary circuit and the estimation of the inventory of radioactivity deposited in it;
- (6) preliminary studies of decontamination efficiency for contaminated graphite;
- (7) analysis of the radiological consequences of burning of most of the irradiated graphite after the deferred removal stage;
- (8) choice of technology for the conditioning of highly contaminated graphite blocks;
- (9) forming the strategy for the decommissioning of the AM reactor and conceptual planning of the decommissioning effort;
- (10) preliminary inventory of cumulative radioactive wastes and contaminated reactor facility components due to reactor operation; and
- (11) estimation of radiological consequences for a hypothetical accident at the AM reactor during the deferral stage as the first step of analyzing its safety during this period.

3. Activity inventory

The radioactive materials inventory for the AM reactor, based on a study using a combination of experimental and calculation methods, has been completed. Computer codes were used to calculate the activation of impurities in the core, reflectors, shield and construction materials based on the actual operation schedule, burnup, buildup and decay of activity. These codes were ORIGEN-2.1 [1] and a recently modified version of the SABINE-3 [2] shielding code, which was named SABINE-3.1. This version using the modern version of ABBN nuclear constants [3] and modified removal cross-sections, can calculate the activation of materials and activation dose rates.

The code SABINE-3.1 has been validated against the IAEA test benchmark [4] for pressure vessel and shield activity of the Japanese JPDR reactor, which had 12 years of operation and 15 years of decay. The code was also validated against the activation photon dose rate distribution in the concrete shield of the WWER-440 reactor in the Armenian NPP [5], which had 10.5 years of operation and 1.5 years of decay. To validate the calculation of activity due to impurities, calculated results were compared to the measured data for the core graphite of the Russian plutonium production reactors I-1 & EI-2, which had 29 and 33 years of operation respectively, and 8 years of decay. The calculation results and the experimental data [6,7] differed, for the most part, by less than 50% as seen in Table III.

Table III. Reactor I-1 & EI-2 core graphite activity, Bq/g

Nuclide	Reactor	Experimental data	Calculation results
^{60}Co	I-1	$5 \times 10^2 - 9.8 \times 10^3$	1.47×10^4
	EI-2	$5 \times 10^2 - 1.1 \times 10^4$	1.10×10^4
^{14}C	I-1	$8 \times 10^5 - 1.9 \times 10^6$	1.1×10^6
	EI-2	$9 \times 10^5 - 1.3 \times 10^6$	7.5×10^5
^{63}Ni	I-1	$2.5 \times 10^2 - 2.2 \times 10^3$	2.8×10^3
^{154}Eu	I-1	$3.7 \times 10^3 - 8.3 \times 10^3$	1.0×10^4
^3H	I-1	$1.0 \times 10^3 - 7.0 \times 10^4$	1.7×10^6

The considerable difference (a factor 25–100) between the experimental and calculation results for tritium could be explained by the calculation methodology, which considered only the generation of tritium and not the leakage. In the calculations of ^{14}C activity two reactions were considered: $^{13}\text{C}(n,\gamma)^{14}\text{C}$ and $^{14}\text{N}(n,p)^{14}\text{C}$, because the stack of reactors I-1 and EI-2 were cooled with nitrogen, which is the main parent of ^{14}C .

Impurity levels for graphite samples from different cells of the core of the AM reactor had been measured by chemical methods (atomic absorption and emission spectroscopy, flame photometry) and radiometric methods (γ - and β -spectrometry, radiochromatography). The results are given in Table IV and compared with data for other graphite reactors used for the production of plutonium and electric power.

Table IV. Impurity levels for graphite by reactor, PPM

Element	AM	I-1, EI-2 [6,7,8]	Magnox [9]	AGR [9]
Li	0.24	0.01–0.05	0.05	0.05
N	240	240	10	10
Cl	25	15	2	4
Ca	15	7	35	25
Co	0.054	0.006	0.02	0.7
Ni	0.07	0.2	1.0	6.0
Ag	0.003	0.26	0.001	0.001
Eu	0.0022	0.002	0.004	0.005
Sm	0.022		0.04	0.05

The difference is obviously due to the various sources of raw material for reactor graphite. One can see that, for most of the impurities the AM reactor value is somewhere in the middle of the others. For a reactor where the graphite is cooled with nitrogen (I-1, EI-2, AM) the amount of nitrogen in the graphite is higher than in the others.

The comparison of inventories of long lived induced activity in the graphite for different graphite reactors is presented in Table V. From the table, one can see that the activity due to impurities in the graphite of the AM reactor has a value that is somewhere in the middle of the data for a number of reactors.

The problem of fission product contamination of the graphite is very important. According to measurements of core graphite, the mean activity of ^{137}Cs after 10 years of decay is about 1.3×10^5 Bq/g for the AM reactor and $\sim 2.0 \times 10^6$ Bq/g for the I-1 and EI-2 reactors [10]. The ^{90}Sr contamination for I-1 and EI-2 is approximately the same as for ^{137}Cs . For the AM reactor this value is $10^2 - 3.0 \times 10^2$ Bq/g. Thus, the AM reactor is somewhat freer of fission product contamination compared with the plutonium production reactors.

Calculations detailing the activity distribution in the graphite stack of the AM reactor (graphite reflector and shield) were performed with the SABINE-3.1 and ORIGEN-2.1 codes. It should be noted that the large dimensions of the graphite stack results in a large non-uniformity of the neutron flux in the graphite. The neutron fluxes in top and bottom of the stack differ by more than three orders of magnitude.

Table V. Long live induced activity in core graphite after 10 years of decay, Bq/g

Nuclide	AM	I-1 [6,7]	EI-2 [6,7,8]	Magnox [9]	AGR [9]
^3H	2.3×10^5 *	$1.0 \times 10^3 - 7.0 \times 10^4$	4.4×10^5	7.7×10^4	7.2×10^4
^{14}C	1.1×10^6	$8.0 \times 10^5 - 1.9 \times 10^6$	$9.1 \times 10^5 - 1.3 \times 10^6$	5.4×10^4	1.8×10^5
^{36}Cl	1.0×10^4		1.3×10^3	6.1×10^2	2.2×10^3
^{41}Ca	3.6×10^2		5.8×10^2	4.7×10^2	1.1×10^3
^{60}Co	3.2×10^4 **	$4.0 \times 10^2 - 1.6 \times 10^3$	$5.0 \times 10^2 - 7.0 \times 10^3$	1.7×10^4	9.3×10^5
^{59}Ni	4.5×10^0		5.8×10^2	6.0×10^1	5.3×10^2
^{63}Ni	7.8×10^2	$2.5 \times 10^2 - 2.2 \times 10^3$	8.2×10^3	8.2×10^3	1.0×10^5
$^{108\text{m}}\text{Ag}$	4.3×10^1		1.0×10^3	1.4×10^1	3.4×10^1
^{152}Eu	0.20		7.0×10^1	1.4×10^2	1.0×10^2
^{154}Eu	8.3×10^2	$3.7 \times 10^3 - 8.3 \times 10^3$	6.2×10^1	3.3×10^3	2.7×10^3

*) – taking into account leakage

**) – taking into account deposit of activation of Fe isotopes (36 ppm), which is $\sim 5\%$

Also, the effect of intensive burnup of the impurities of Eu and Sm in the core and reflector compared to the shield should be taken into account. The dependence of Eu isotopes activities upon the neutron flux density (content of Eu is 1 ppm, Sm – 10 ppm) after 50 years of irradiation is shown in Figure 2. As a result, activities of Eu isotopes in the core are significantly less than those in the shield. The results of calculations for induced activity in the graphite stack (50.8 tons) and an estimation of the quantities of ^{137}Cs and ^{90}Sr present 10 and 90 years after the shutdown of the AM reactor are presented in Table VI.

The main activities produced in the reactor vessel are isotopes of Ni and Co. Calculations have shown that the highest radioactivity level occurs in the lower part of the vessel ($\delta = 8$ mm). The calculated activity inventory for the reactor vessel and the internal wall of the water bioshield tank ($\delta = 12$ mm) for two decay times are presented in Table VII.

The inventory of activity in the bottom base plate under the reactor is approximately the same. Its surface contamination is not known at this time.

The radioactivity in the external wall of the bioshield water tank is less than the radioactivity in the internal wall by about four orders of magnitude. The radioactivity in the cast iron blocks of the top shield is less than the activities reported in Table VII by two orders of magnitude.

The calculated average long lived activity in the top part of the concrete base of the reactor ($\delta = 60$ cm) is presented in Table VIII.

The induced activity at the surface of the concrete base is ~ 5 times greater than that in Table VIII.

The calculated activity at the surface of the concrete shield beside the reactor is less than the data of Table VIII by 150 times (negligible), and for ^{60}Co is $\sim 10^2$ Bq/g, and for ^{152}Eu ~ 0.2 Bq/g. These results are very similar to the results of activity measurements in samples of this concrete.

The results of the activity inventory can then be used to analyse the radiation doses to be incurred during the dismantling of the reactor.

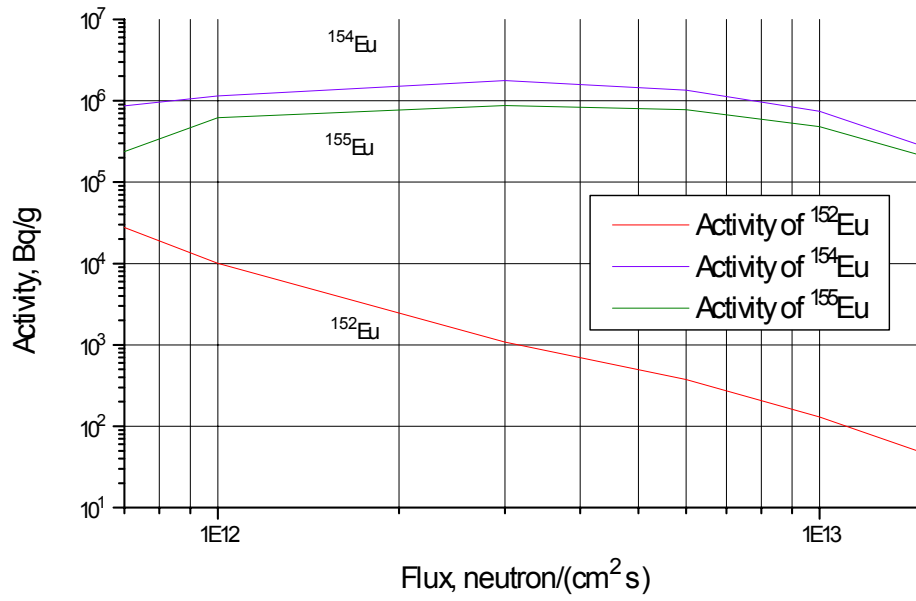


Figure 2. Dependence of EU isotopes activities from neutron flux.

Table VI. Integral radioactivity inventory of AM reactor graphite stack, Bq

Nuclide	10 years of decay	90 years of decay
³ H	7.6×10 ¹²	8.5×10 ¹⁰
¹⁴ C	3.7×10 ¹³	3.7×10 ¹³
³⁶ Cl	3.5×10 ¹¹	3.5×10 ¹¹
⁴¹ Ca	1.2×10 ¹⁰	1.2×10 ¹⁰
⁵⁵ Fe	2.6×10 ¹¹	-
⁶⁰ Co	2.7×10 ¹¹	7.3×10 ⁶
⁵⁹ Ni	1.5×10 ⁸	1.5×10 ⁸
⁶³ Ni	2.6×10 ¹⁰	1.4×10 ¹⁰
^{108m} Ag	1.5×10 ⁹	9.7×10 ⁸
¹³⁴ Cs	2.3×10 ¹⁰	-
¹⁵² Eu	3.3×10 ⁸	5.6×10 ⁶
¹⁵⁴ Eu	6.2×10 ¹⁰	9.8×10 ⁷
¹⁵⁵ Eu	4.9×10 ¹⁰	-
¹³⁷ Cs	3.2×10 ¹²	5.1×10 ¹¹
⁹⁰ Sr	6.1×10 ⁹	8.6×10 ⁸

Table VII. Activity inventory in lower part of reactor vessel and internal wall of bioshield water tank

Component	Nuclides	Decay 10 years, Bq/g	Decay 90 years, Bq/g
Reactor vessel	⁵⁵ Fe	7.5×10^7	-
	⁶⁰ Co	2.0×10^7	5.3×10^2
	⁵⁹ Ni	8.2×10^3	8.2×10^3
	⁶³ Ni	9.6×10^5	5.3×10^5
Bioshield Tank	⁵⁵ Fe	3.9×10^7	-
	⁶⁰ Co	1.0×10^7	2.7×10^2
	⁵⁹ Ni	4.3×10^3	4.3×10^3
	⁶³ Ni	5.1×10^5	2.7×10^5

Table VIII. Average activity in the concrete base, Bq/g

Isotope	Decay 10 years	Decay 90 years
⁴¹ Ca	2.2×10^3	2.2×10^3
⁶⁰ Co	4.2×10^3	1.2×10^{-1}
¹⁵² Eu	1.5×10^1	2.6×10^{-1}
¹⁵⁴ Eu	1.3×10^1	2.1×10^{-2}

4. Dismantling analysis

One of the dose rate measurements at the top boundary of the unloaded core of the AM reactor after 1 year of decay was about 1 mSv per second. Calculation results for dose rates at the same place after different decay times are presented in Table IX.

Dose rates from other parts of the graphite stack, except for the lower part of the reflector, are significantly less.

After a long period of decay, the radiation fields during the cutting of the reactor vessel result from radioactivity in the reactor vessel itself, with only a small fraction (~ 5%) coming from activity in the core and impurities in the bottom graphite reflector. The reactor vessel largely blocks the latter. After 90 years of decay the dose rate will be about 100 μ Sv/h at a distance of 50 cm from the reactor vessel.

Table IX. Dose rate at the top boundary of the unloaded am core, μ Sv/h

Decay, years	Dose Rate
30	7.25
50	4.10
70	2.54
90	1.62
110	1.02

It should be noted that this dose rate exceeds the allowed level for the normal operation (10 $\mu\text{Sv/hour}$). Thus, one will have to apply local shielding or limit the working time for operators and increase their number.

To evaluate the total dose due to the cutting of the reactor vessel using a mechanical method, the following assumptions are made. The reactor vessel will be cut into 5 rings and each ring cut into 11 fragments for a total of 55 fragments. The cutting of a single fragment takes about 3 hours, preparation and positioning operations take about 1 hour per fragment. To perform the same cutting using a thermal method requires about 3–6 min for cutting and 30 min for preparation and positioning operations per fragment. The resulting dose for personnel without remote operated equipment is presented in Table X. The dose incurred when cutting the bioshield water tank is less than that due to cutting the reactor vessel by a factor of two.

Table X. Dose due to cutting of reactor vessel (90 years of decay)

Method	Time of external irradiation, Hour	Dose of external irradiation, Man·mSv	Dose due to inhalation, Man·mSv	Summary, Man·mSv
Mechanical	210	12.3	2.6	14.9
Thermal	33	2.0	1.1	3.1

To estimate the inhalation doses via calculations it was assumed that the dismantling staff would wear respirators when using the thermal method of cutting the reactor vessel. As a result, the thermal method of cutting is more suitable and doses are significantly lower than the allowed annual dose limit of 20 mSv. After filtering and being exhausted through 100 m of ventilating pipe, the radioactivity released to the environment due to the dismantling of the reactor vessel is negligible.

To discuss the dose implications of the dismantling of the graphite stack, one must first understand its construction. Each graphite block is a hexahedral prism with an inradius of 12 cm and one of a number of different heights: 60, 45 and 30 cm. The central part of the stack containing the core and lower reflector, consists of 151 columns of 4 blocks each. During analysis of the doses during manual dismantling (without use of remote operated manipulators) of this part (after dismantling of reactor vessel and other parts of the graphite stack) the doses were calculated for different decay times – 70 and 90 years. The results were 430 man·Sv and 274 man·Sv respectively. These doses are not acceptable, because they equate to a annual dose limit for 15–20 operators during the dismantling period. However, the use of local steel shields with a thickness of 10 cm can decrease these doses by more than 100 times. So, dismantling of the graphite stack of the AM reactor is feasible after 70–90 years of decay. The doses from the dismantling of cast iron shield components are not significant.

When dismantling the graphite stack the graphite blocks should be sorted based on their activity and contamination levels. After conditioning of the blocks they should be committed to a special depository for decay. To provide safe and durable isolation of the graphite blocks from the environment they should be infiltrated with a of special polymeric conservation agent [11].

5. Graphite burning

To minimize the amount of graphite to be burned, a special analysis of radiological consequences of burning of the “pure” part of the graphite stack was carried out. According to calculation and experimental estimations, 97% of the graphite stack is not contaminated with fission products and can be considered as “pure”. The radiological consequences of burning the irradiated reactor graphite were analyzed for a 15 km area around Obninsk. The distance between Obninsk and the reactor site is about 1 km.

The area contains about 44 % agricultural lands whose main agriproducts are potatoes, vegetables, fruits and white straw crops. Products from dairy and cattle farming in the area are meat, milk and eggs.

The population of the area is about 200,000, mainly in Obninsk (100,000) and 7 towns and villages (with populations from 3,000 to 30,000). For the analysis it was conservatively assumed that the population eat imported bread products, but consume other foods which are produced from the native agriproducts

During the burning of graphite the radioactive isotope ^{14}C is released to the atmosphere in the form of CO_2 . This gas is absorbed by air breathing plants during photosynthesis, after which it makes its way to the human body by the ingestion pathway (through vegetable or meat-milk food chains). The dose due to the inhalation of ^{14}C is negligible.

The analysis of the radiological consequences of food intake used the official market-basket data, which is an average for the Russian Federation at present time (see Table XI).

The burning of the pure parts of the graphite (49.5 tons), following 90 years of decay, will release $\sim 4 \times 10^{13}$ Bq of ^{14}C (see Table VI). This release was analyzed by applying the recommended IAEA [12] Gauss model of atmospheric diffusion to the agricultural lands in the area around Obninsk. The result is a conservative calculation of equivalent doses due to the consumption of contaminated food (taking into account the negligible effect of nuclides from Table VI other than ^{137}Cs and ^{90}Sr) equal :

- For adults – 1.4 μSv per year
- For pensioners – 1.1 μSv per year
- For children – 2.0 μSv per year.

Comparing these results with the IAEA and Russian Federation legislated dose levels [13, 14] of 1 mSv/year for members of the public, it can be seen that the radiological effect from burning of the pure parts of the reactor AM graphite is about 0.1 or 0.2 % of the allowed annual dose, i.e., it is negligible.

Table XI. Average food intake**, kg per year

Kind of product	Adults	Pensioners	Children
Vegetable products	230	200	267
Meat & eggs	43	29	41
Milk	211	200	296

** This is the average for the Russian Federation without bread and bread products.

It is very interesting to compare these results with the amount of natural ^{14}C in the human body. According to UNSCEAR publication [15], the average effective equivalent dose due to natural ^{14}C in the human body is about 13 μSv per year. Thus, according to the conservative evaluation, burning of the AM reactor graphite can increase amount of ^{14}C in human body about 10 or 15 %. Since the actual native food consumption is $\sim 30\%$, the value equals only $\sim 3\text{--}5\%$. This is within the data accuracy of the UNSCEAR publication. Thus, the radiation risk from burning the AM reactor graphite is negligible.

6. Contaminated graphite

One option that should be considered for the safe storage of the graphite blocks is to decontaminate the surfaces contaminated with fission products. All of these blocks, which number about 100, are from the core and lower reflector. Preliminary experiments indicated that surface contamination of graphite with fission products could be partially removed by a concentrated solution of nitric acid, decreasing the surface activity of ^{137}Cs at the same time.

The analysis of graphite probes from the reactor core resulted in a preliminary estimate that there are some dozens of the high level contaminated blocks, based on the evaluation that the mass of ^{235}U spilled was 4 g. For these blocks it is recommended to stabilize the accumulated radioactivity by the method of high-temperature self-spreading synthesis in stable carbide-oxidic composite material [16].

7. Safety during storage of reactor

The list of initiating events that could lead to possible accidents during the long deferral period needs to be developed as the basis for performing a total safety analysis of the unloaded AM reactor. Once the list is obtained an analysis of the radiological consequences of each of them results in the impact on the staff and the public. It is time-consuming and expensive work, which could include the performing of experiments with reactor graphite samples under extreme conditions (high temperature, solubility in ground water etc).

To estimate the scale of radiological consequences during the preliminary stage of safety analysis for the long decay period, an analysis was carried out for one hypothetical accident involving a near ground release of 23 kg of graphite paste (about 10% of the full amount, used for repair of graphite stack of reactor AM in 1971 and 1987) after 50 years of decay. The chemical content of the paste was measured via activation analysis. The main nuclides and their activity were: ^3H - $4.4 \times 10^9 \text{ Bq}$; ^{14}C - $5.0 \times 10^{10} \text{ Bq}$; ^{36}Cl - $3.6 \times 10^8 \text{ Bq}$.

The maximum calculated effective equivalent dose per year due to the accident, at a distance of more than 1 km from the reactor, is 0.5 mSv/year. That is less than the dose limit for the public for a year, 1 mSv. This maximum occurs near the reactor and it is 14 mSv/year. This is less than the dose limit for Staff of Group A (operating staff – 20 mSv/year), but it exceeds the dose limit for Staff of Group B (assistant staff – 5 mSv/year). So, an accident of a similar scale would require decontamination of the terrain near the reactor block, but it is not a catastrophe.

8. Radioactivity inventory for radwaste storage of hot cell

The AM reactor was the first research reactor in Russia. The reactor building contained a hot cell, employed in the cutting up of spent fuel assemblies for the subsequent investigations. During its long period of operation the accumulation of radioactive wastes,

stored under the hot cell, amounted to 21.5 tons of irradiated graphite sleeves of fuel assemblies and control rods, and 7.5 tons of irradiated steel fragments of fuel assemblies. Measurements of activity for a “fresh” irradiated graphite sleeve allows one to identify the aggregate activity of radwastes in storage. The results of the estimation are present in Table XII.

Comparing Table XII and Table VI, it can be seen, that the total activity of the graphite stack for each nuclide other than ^{63}Ni from stainless steel, is significantly greater (by factors of 10 to 1000 times) than the activity of radwastes in storage at the hot cell. So, the decision regarding the long lived period of decay for the AM reactor without dismantling also applies to the wastes stored at the hot cell regarding unloading them into a central radwastes storage. One is advised to follow this decision in order to decrease the dose to the operating staff due to the unloading of the hot cell radwastes storage.

9. Primary circuit

The activity of surface depositions in the primary circuit is due to the activation of corrosion products and, for ^{60}Co , is about $\sim 10^8 \text{ Bq/m}^2$. Experiments were performed on deep decontamination for fragments of the primary circuit (surface contamination $\sim 10^7 \text{ Bq/m}^2$) with concentrated hot solutions of alkali and nitric acid. They showed the ability to decrease the surface activity ~ 100 times. The estimated volume of liquid radioactive wastes from deep decontamination of the primary circuit is about $60\text{--}70 \text{ m}^3$, estimated radioactivity of ^{60}Co is about 10^{13} Bq (after 5–10 years of decay). A more effective and natural method of decreasing the activity is decay time. A simple estimation shows, that the unconditional clearance level (for unrestricted use) for surface contamination (10 Bq/cm^2 for ^{60}Co) recommended by IAEA [17] will be achieved after 50 years of decay. The calculated estimate of ^{63}Ni surface contamination at this time is less than 10^3 Bq/cm^2 , i.e., less than 5 % of the unrestricted clearance level for that nuclide for construction materials.

Table XII. Integral long live radioactivity inventory for radwastes storage of hot cell, Bq

Material	Nuclide	10 years of decay	90 years of decay
Graphite	^3H	1.6×10^9	1.8×10^7
	^{14}C	4.7×10^{10}	4.7×10^{10}
	^{36}Cl	6.7×10^7	6.7×10^7
	^{60}Co	4.3×10^8	1.2×10^4
	^{63}Ni	4.1×10^8	2.2×10^8
	^{152}Eu	1.1×10^9	1.9×10^7
	^{154}Eu	5.5×10^9	8.7×10^6
	^{137}Cs	1.1×10^{11}	1.7×10^{10}
	^{90}Sr	1.6×10^9	2.3×10^8
Stainless steel	^{60}Co	6.6×10^{12}	1.8×10^8
	^{63}Ni	1.2×10^{14}	6.6×10^{13}

10. Decommissioning plan

The preliminary planning schedule for the decommissioning of the AM reactor includes the following:

- (1) Preliminary inventory of the cumulative long lived radioactivity in the AM reactor and development of the decommissioning strategy – 1998–2000.
- (2) Development of Decommissioning Project – 2001–2003.
- (3) Safety Analysis for the deferral stage of the defuelled AM reactor – 2004.
- (4) Final shutdown of the AM reactor – December of 2004.
- (5) Cooldown and defuelling of the reactor, drainage of coolant – 2005.
- (6) The first deferral stage – 2006–2010.
- (7) Complex Engineering and Radiation Survey of the reactor and the storage of radioactive wastes – 2007–2009.
- (8) Updating of the Decommissioning Planning for the AM reactor, including revisions to the safety analysis for the second deferral stage.
- (9) Isolation of the reactor. Decontamination of reactor components in zones needed for maintenance during the deferral stage – 2011–2013
- (10) Removal of the radioactive wastes in storage or its isolation for safety.
- (11) Monitored deferral stage – 2013–2095.
- (12) Dismantling and removal of some stainless steel components of the primary cooling circuit – 2055.
- (13) Detailed Engineering and Radiation Survey of the reactor and all components still on-site to develop solutions for dismantling them.
- (14) Dismantling of reactor and reactor facility (if required) – 2095–2100.

11. Summary

The conceptual decommissioning strategy for the AM reactor consists of long term safe storage with deferred dismantling without the use of remotely operated equipment. Isolation of the shutdown reactor from the surrounding environment will be provided. Nonetheless, during the 90 year deferral period decommissioning solutions will be considered only after careful engineering investigations of the isolating barriers (including the reactor vessel) and the reactor building up that date.

After that one of two different solutions will be chosen:

- dismantling of the graphite stack and transportation of the graphite blocks to a special depository; or
- continued storage of the graphite in the reactor for the next 50–100 years, after reinforcing the reactor vessel and other isolating barriers.

During dismantling of the graphite stack, the graphite blocks should be carefully sorted based on their level of contamination. Highly contaminated blocks, which also contain high levels of induced activity, should be reprocessed with a special technology. This technology includes the grinding of the graphite blocks and their transformation into stable carbide-oxide composite material under high temperatures.

The amount of stored graphite could also be decreased by burning the uncontaminated parts of it. Radiological consequences of this burning for the exposed population are acceptable. However, the final decision on the technology to be used for disposal of the graphite will be made in the future, after a deferral period.

After 50 years of decay, steel reactor components, contaminated with activated corrosion products, could be reused without decontamination or any other limitations or, in the case of necessity, after 25 years of decay and careful decontamination.

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Decommissioning of nuclear installations at CIEMAT

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Abstract. This report presents the work carried out by CIEMAT in the frame of decommissioning the research reactor JEN-1. Studies for evaluating different metal cutting techniques, including plasma-arc cutting, contact-arc cutting and mechanical saw cutting led to assessing the performance, advantages and associated problems for each technique. The main metallic material studied was aluminium, but some experiments with stainless steel were also conducted. Melting was also studied as a decontamination technique and as a way to reduce volume and facilitate the management of radioactive waste.

1. Introduction

Spain has a long history in nuclear energy. In the early fifties the Junta de Energía Nuclear (JEN) was created, the institution from which the CIEMAT later evolved. JEN was a public institution under the Ministry of Industry and was created + to carry out research and development on the peaceful uses of nuclear energy. JEN was transformed into CIEMAT ("Centro de Investigaciones Energéticas Medioambientales y Tecnológicas": Energy, Environmental and Technological Research Centre) in 1986.

Among the more important installations and equipment that JEN acquired as it developed was the JEN-1 Reactor. JEN-1 was an experimental, pool type, research reactor, moderated and cooled by light water, with a thermal power of 3 MW that first went critical in 1958. The reactor operated almost continuously from 1958 to 1984, with a total generated energy of 2700 MWD. The normal core load was 30 MTR (Material Test Reactor) type fuel assemblies, containing 4.05 kg of U-235, producing an average thermal flux of $2.2 \times 10^{13} \text{ n} \cdot \text{cm}^{-2} \cdot \text{s}^{-1}$. It was upgraded in 1969 and again in 1984 with important modifications to the control system. The order for its final shutdown was given in 1987, after which the decommissioning process for the reactor was begun.

There are other nuclear facilities present in CIEMAT which, like the JEN-1 Experimental Reactor, are included in a general decommissioning plan for CIEMAT: the M-1 Plant (reprocessing pilot plant), a radioactive liquid waste storage facility, a facility for the conditioning of radioactive liquid wastes, etc. This plan is called the "Integral Plan to Improve CIEMAT Installations (PIMIC)" that has been established and started up recently. Section 9 presents further information about this Plan.

2. Objectives of the project

The main objectives of the project "Decommissioning of nuclear installations at CIEMAT" include the following:

- Evaluation of underwater cutting of metals by mechanical (pneumatically actuated saw) and thermal (plasma arc, CAMC, etc.) techniques.
- Studies on melting of metallic materials, resulting from the underwater cutting, with the aim of volume reduction and/or free release.

The largest fraction of the material used in these studies came from aluminium components from the JEN-1 Reactor.

3. Initial activities

Before carrying out decommissioning tasks, prerequisite activities had to be carried out in the JEN-1 reactor installation. These activities were:

- Handling and transferring 40 spent fuel elements. These elements were removed from the reactor pool and stored in wells in order to carry out the preliminary dismantling activities. Later on, these spent fuel elements were loaded into GOSLAR casks and transported abroad for reprocessing.
- The underwater dismantling of reactor internals began once the fuel assemblies were removed from the JEN-1 pool. Dismantling was achieved manually by operating from one of the two moveable JEN-1 pool bridges. Work was performed at the 8 m depth by means of elongated and articulated poles. These operations were controlled and monitored via TV cameras.
- Studies and assessments for planning and designing the cutting and melting facilities.
- Building the facilities.
- The licensing process for these facilities.

In accordance with the decommissioning program, components from the high power zone of the JEN-1 (the core, control mechanisms, irradiation devices, etc.) and an important part of the low power and storage zone components were successfully removed. The two cooling towers from secondary circuit were also completely dismantled.

4. Facilities description.

4.1. Underwater cutting facility

An underwater cutting facility was designed, built and installed into a confined area of the JEN-1 pool, to be used in the application of different cutting techniques for radioactive metals. The most important features of this facility are:

- a handling and positioning system;
- a cutting basin;
- a purification system for the water in the cutting basin;
- an exhaust system for gases and aerosols; and
- a control system for the cutting tools.

The handling system has five degrees of freedom: three linear movements and two rotational ones. The "swivel and pitch" device, which performs the two rotational movements, is located at the lower end of the vertical mast. This device carries the different cutting tools installed for the cutting technique studied.

The stainless steel cutting basin is a cylindrical structure, 2.85 m in diameter and 5 m high; with a capacity of 32 m³. The basin is immersed in the reactor pool, and fastened to the pool walls. The usual working depth is 3 to 4 m underwater.

A water purification system exclusively dedicated to the water in the cutting basin is used to keep turbidity and salts content as low as reasonably possible. It consists of a sand

filter, ion-exchange resin bed and several “in depth cartridge” filters, besides a pump, instrumentation and pipes.

A hood over the basin collects gases arising from the cutting operations. Gases are conducted through different filter systems by an extractor. The hood is equipped with hydrogen measurement equipment and also two isokinetic aerosol samplers have been installed on the pipes.

Figure 1 presents a general view of this cutting facility. The cutting operation is managed from a Control Room, located close to the top of JEN-1 pool, which contains the controls for the positioning system, the remote master robot control, and a TV remote monitoring system.

A master/slave robot manipulator, Manipulator Handling System, abbreviated MHS, designed and built by LENTJES MCE Anlagen und Rohrleitungsbau GmbH (Austria) has also been used. This remotely operated manipulator robot was acquired by CIEMAT as part of an R&D project co-ordinated by UWT-Hannover, Hannover University (Germany). The MHS has 7×2 (slave/master) rotational movement axis, and consists of manipulators, electric systems, and hardware and software controls. It was used for experiments with a Contact Arc Metals Cutting (CAMC) technique (see later). This manipulator, working in the cutting vessel, can be seen in Figure 2.

4.2. Melting facility

One objective of this project was the study of a melting process as a decontamination technique. It has two main purposes: recycling material and volume reduction. An experimental melting facility for low level radioactive materials has been designed and built close to the JEN-1 reactor. The melting facility is provided with:

- a furnace system;
- an exhaust system for gases and aerosols; and
- auxiliary systems (purging gas, fire fighting equipment, etc.).

The operating license of this facility, issued by the Spanish regulatory authority (CSN), allows the melting of radioactive materials with an average specific activity of up to 500 Bq/g. Experiments with higher activity require a specific approval from the CSN.

The melting facility is provided with auxiliary equipment for monitoring radiation levels and airborne particles at the working area and in the furnace cubicle.

The melting system includes:

- Induction furnace with a ceramic crucible (capacity for aluminium melting: 20 kg) and a hydraulic device for upsetting the furnace;
- Thyristor static generator (nominal power: 50 kW; frequency: 3.000 Hz), electric systems and controls.
- Water cooling system. Two cooling loops are employed: the primary circuit directly applied to the furnace coil and the generator, and the secondary circuit with a cooling tower (capacity: 50,000 kcal/h).

A partial view of this melting facility can be seen in Figure 3.

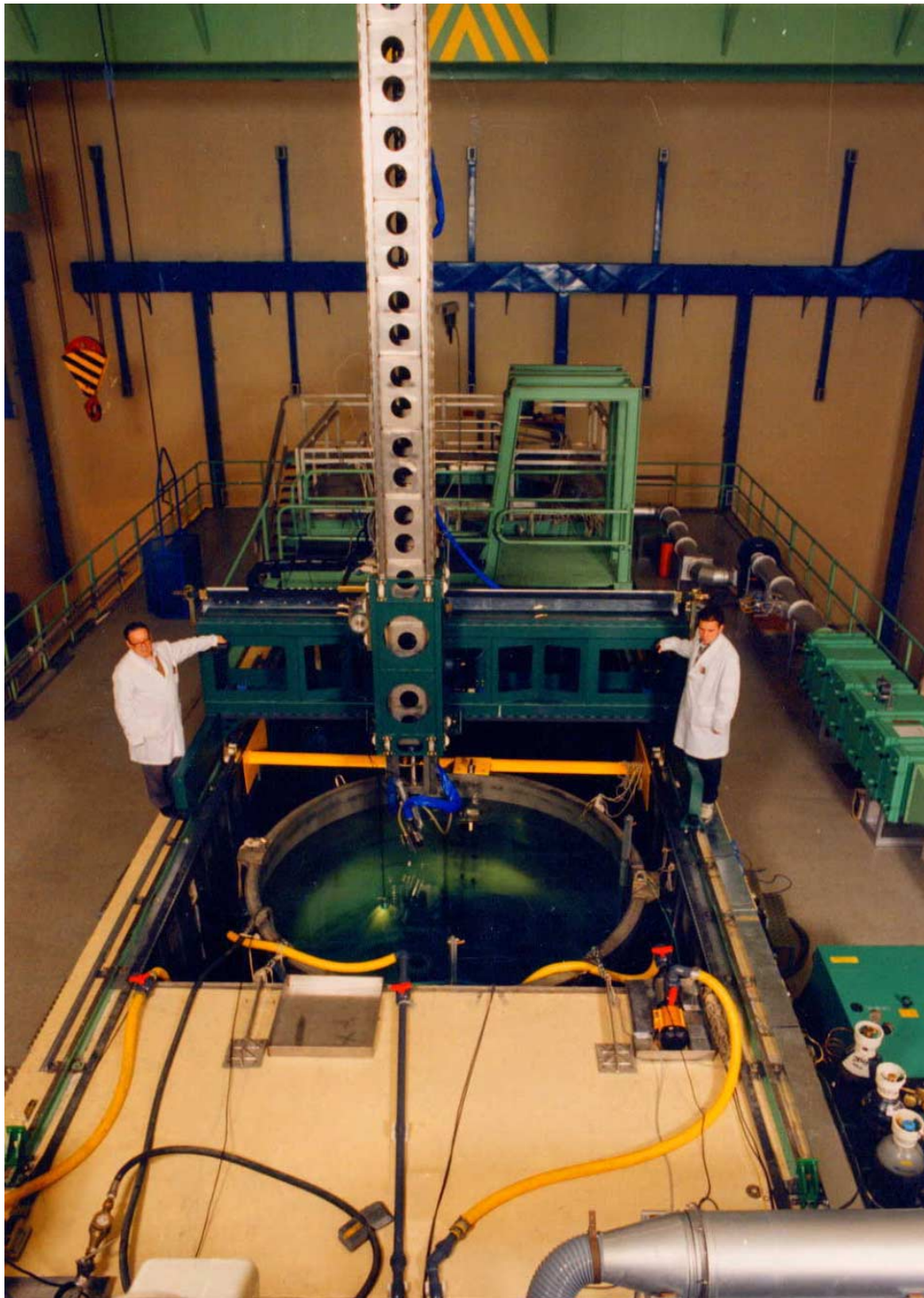


Figure 1. Underwater cutting facility.

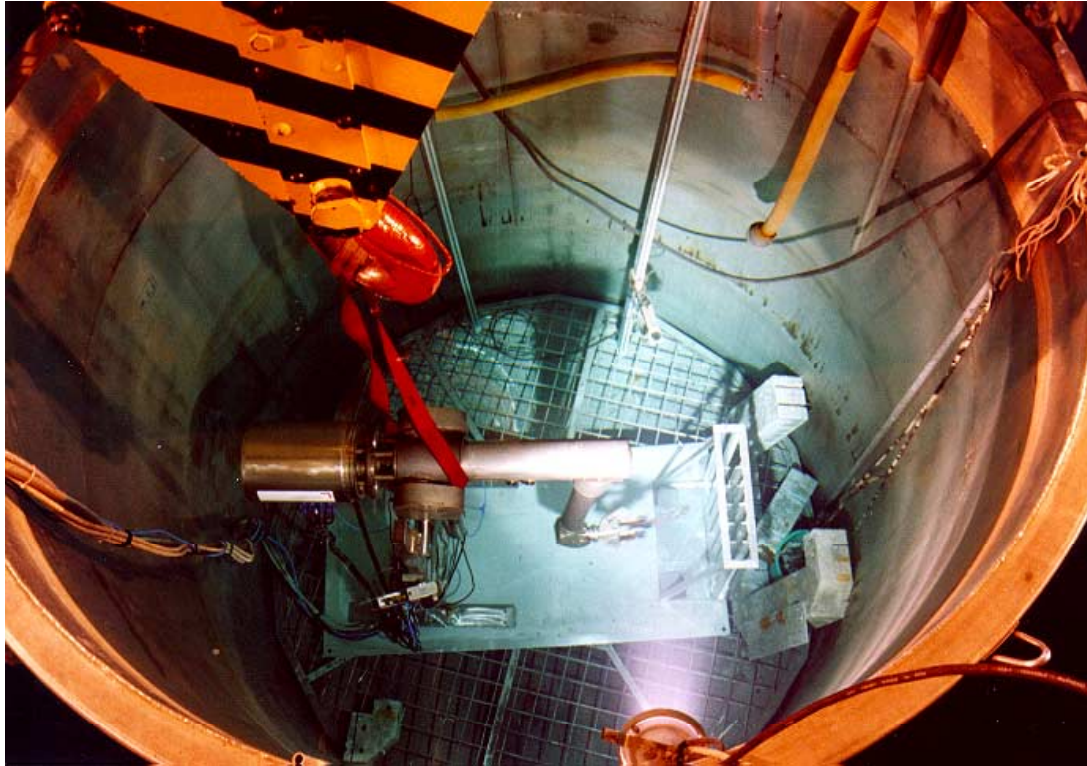


Figure 2. Rrobotic manipulator.



Figure 3. Melting facility.

5. Cutting techniques used

5.1. Cutting by plasma-arc

Plasma-arc cutting is a thermal cutting method. An electric high-voltage arc between an auxiliary electrode and a torch is established inside the torch (“pilot arc”). A gas, nitrogen in this case, flowing through the torch is ionised, producing a plasma gas due to the electric arc. This plasma dart is transferred and focused to the positive electrode, the piece to be cut or working piece, from the torch. This plasma gas reaches temperatures higher than 15000°C in a small dart shape, which, focused in a small area of the working piece, melts the material. Melted material is directly taken out of the kerf by a second high pressurised gas flow (usually CO₂), which also cools the electrode. The plasma torch electrode is usually made from titanium and is internally water cooled.

The plasma-arc cutting tool is easily fastened to the five axis handling system (see Figure 4) due to its light weight. Flexible connections are used to connect the tool to the high-powered electric source (100 kW, 500 A c.c.), the high pressure gas sources and the cooling water pump. Thus, the positioning device can be easily moved along the working piece, guiding the tool to the selected cutting points.

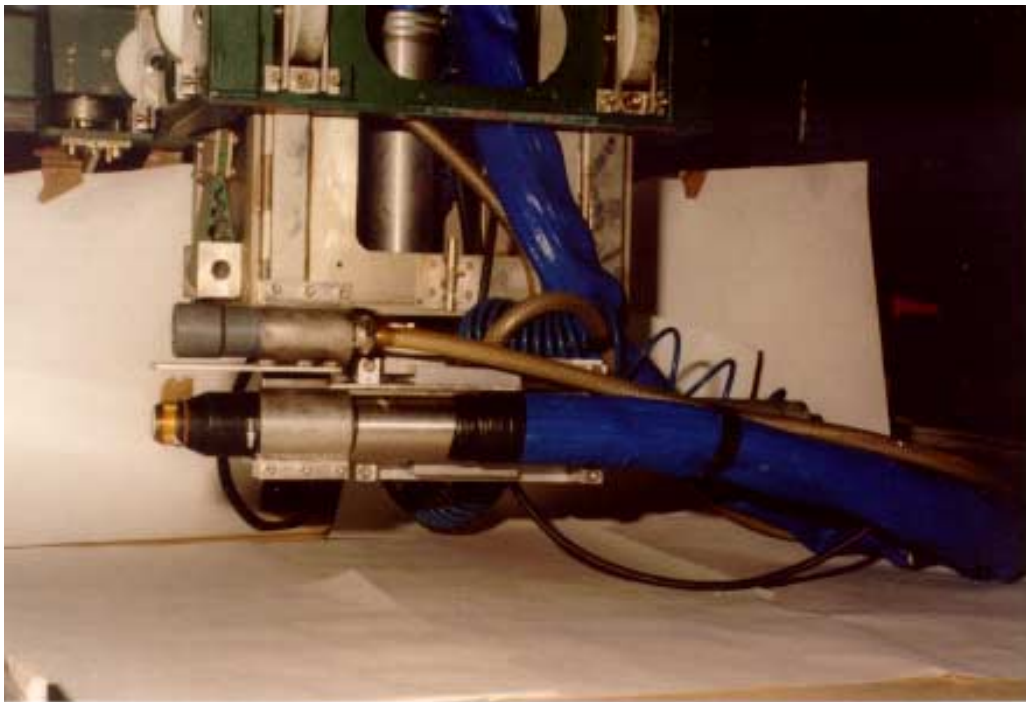


Figure 4. Plasma-arc cutting torch attached to the 5 axis handling system.

5.2. Cutting by CAMC

As mentioned in 4.1 the experiments with this cutting technique were carried out under a joint project with CIEMAT (Spain), UWT-Hannover, Hannover University (Germany) and LENTJES MCE (Austria).

The CAMC system is a thermal metal cutting application that is easy to use. CAMC consists of applying a high-voltage directly between a permanent (“non-consumable”) electrode and a metal working piece (positive pole). A high voltage arc is established between the permanent electrode, made of pressed graphite plates, and the working piece. This high

energy produces local melting of material, which is immediately taken out of the kerf by a high-pressure water jet.

The electric power source provides 24–50 V DC at 200–2000 A. The wires from the electrical power source are cooled by an internal forced flow of water. An 8 m³/h flow high pressure pump is used for both cooling circuits: wire cooling and the water jet for kerf-electrode rinsing. The cutting tool and positioning system must be electrically isolated to avoid discharges.

This cutting tool has been used with two different positioning systems: the five axes device and a master-slave robot designed by LENTJES MCE known as "manipulator handling system" (MHS). The CAMC cutting tool has been designed with interfaces adapted to both positioning systems.

5.3. Mechanical cutting

This is a group of cutting techniques belonging to the “fragmentation” type. The equipment used in CIEMAT to carry out experiments in mechanical cutting is a pneumatically actuated commercial reciprocating saw. The saw has a stroke of 60 mm at 330 strokes per minute. It has been modified and adapted to be capable of operating under watertight conditions and is also provided with a pneumatic system to hold it against the piece to be cut and to control the saw blade pressure against the piece. Each of these pneumatic systems has its own supply. The saw can cut pipes up to 200 mm in diameter (or other material of like thickness). The blades used are 400 mm long, 2 mm thick with 8 teeth per inch.

Figure 5 shows the saw and the pneumatic control system working on a mock-up.

6. Materials studied

6.1. Selected components

Certain components of the JEN-1 reactor were selected for this project. The selected components were all made from aluminium, except for the guide tubes for the control blades, which were stainless steel. Two racks for storing fuel elements from the JEN-1 reactor and some plates coming from storage racks of the Almaraz Nuclear Power Plant were also selected. The following is the list of selected radioactive components:

- Core grid
- Grid support
- Control blade housings
- Ionization chambers support
- Primary cooling circuit plenum
- Guide-tubes of control blades
- 5 + 1 fuel elements rack
- 18 fuel elements rack
- Almaraz NPP fuel element storage rack

Non-radioactive materials were also used for training in cutting with the CAMC technique such as full size mock-ups of:

- 5 + 1 Fuel elements rack
- Double-T beam grid support
- Guide-tubes of control blades



Figure 5. Mechanical saw working on a mock-up.

All of these components are described below.

The core grid ("Al-Mg3"; Al - 97%, Mg - 3%-) was a rectangular frame (1180 mm x 930 mm, by 150 mm thick) with chamfers at its corners. It had a network of 81 (9x9) square cells (each 67,7 mm x 67,7 mm), assembled from 10 mm thick aluminium plates.

The grid support (Al-Mg3), anchored at the bottom of the pool in the high-power position, was a framework in a pyramidal trunk shape with a rectangular cross section. It was made from four vertical beams and four horizontal ones, which form the lower base. All beams had a double-T profile (120 mm x 100 mm, thickness 20 mm). The upper part of this support was a frame with shape, dimensions and holes suitable to support and hold the core grid structure. The overall dimensions of this frame were 1725 mm × 1475 mm (lower base) and 1900 mm in height.

The control blade housings ("Al-99.5", Al – 99.5 %) were made from two 965 mm high by 716 mm wide by 3 mm thick plates held 19 mm apart by three solid separators, two at the ends (60 mm wide) and one at the middle (10 mm wide). The control blades could slide in between.

The ionization chambers support (Al-99.5) was made from an assembly of structures and plates. Its overall dimensions were 1470 mm × 630 mm × 1180 mm (l × w × h).

The primary cooling circuit plenum (Al-Mg3) was a 10 mm-thick aluminium box, 820 mm × 770 mm × 1165 mm (l × w × h). It had a lateral sliding gate on the top, and was

joined directly to the core grid structure. It also had a cylindrical tube at the bottom, which connected to the primary cooling pipe.

The components mentioned above are represented in Figure 6. Other components used were:

- the 5 + 1 fuel elements rack (Al-99.5) made from an assembly of L-profiles ($30 \times 30 \times 3$ mm), with overall dimensions of $353 \times 662 \times 900$ mm ($l \times w \times h$);
- the 18 fuel element rack (Al 99.5) made from an assembly of T-profiles ($25 \times 15 \times 3$ mm) and L-profiles ($20 \times 20 \times 3$ mm), with overall dimensions of $353 \times 1560 \times 900$ mm ($l \times w \times h$);
- the guide-tubes for the control blades, an assembly of four stainless steel tubes 84 mm diameter and 2 mm thickness, with overall dimensions of $286 \text{ mm} \times 258 \text{ mm} \times 8877$ mm ($l \times w \times h$); and
- the stainless steel Almaraz NPP fuel element rack plates. $113 \times 416 \times 6$ mm.

The mock-ups for training in the CAMC technique, shown in Figure 7, were full-scale constructions from the same materials as their originals.

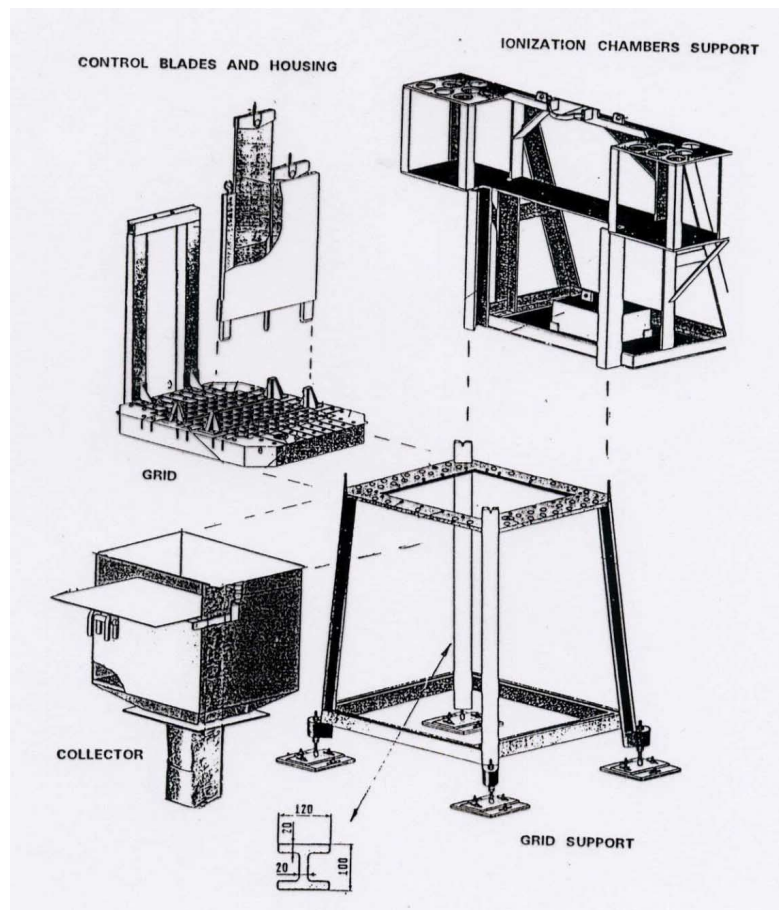


Figure 6. Some selected components.

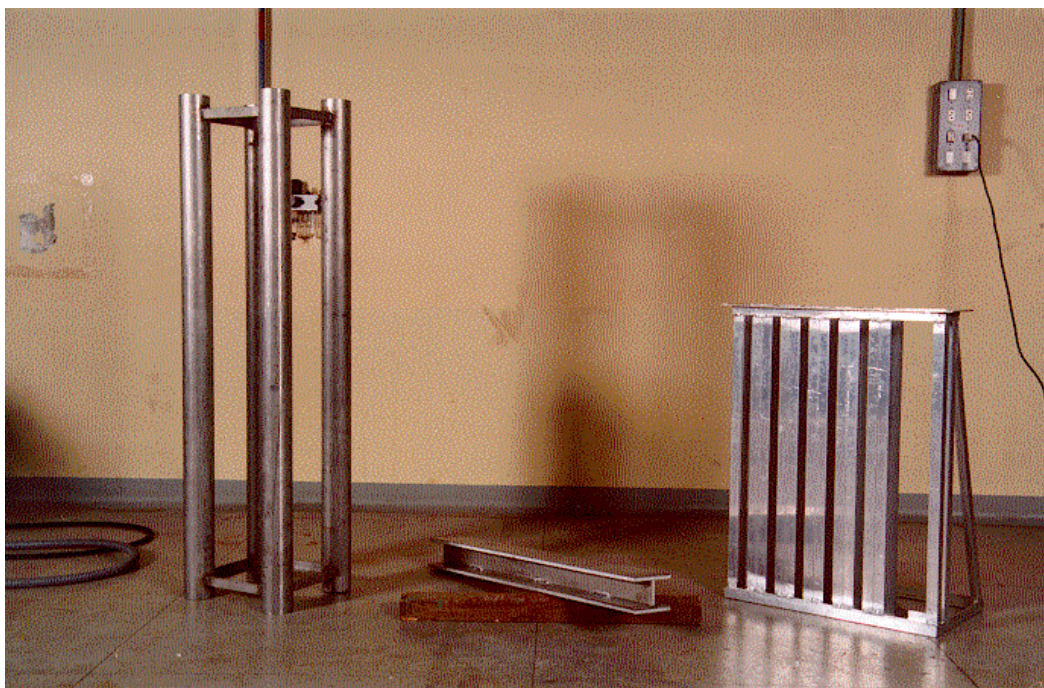


Figure 7. Mock-ups for inactive cutting tests.

6.2. Characterization of materials

The above mentioned reactor components were raised in the JEN-1 pool after dismantling all reactor internals. These components were kept 1.5 m below the pool surface in order to keep the dose rate at the water surface lower than 2 $\mu\text{Sv/h}$. At this position the components were sampled via core drilling. The radioactive characterisation was achieved by α , β and γ spectrometry of samples taken from these pieces and the dose rate was measured directly.

The results obtained showed that the main radionuclides present, with the highest specific activities, were Fe-55, Co-60 and Cs-137, with some Eu-152 on the control blade housing. These results are shown on Table I.

7. Results of experimentation

7.1. First phase (preliminary work)

The first phase includes cutting experiments by plasma arc and mechanical saw on several aluminium components from the JEN-1 core:

- primary cooling circuit plenum;
- a plate from the core grid;
- two control blade housings; and
- ionization chambers support

These experiments garnered basic information on the cutting techniques as applied to aluminium materials. They explored such aspects as: basic cutting parameters (speed, gases or water pressure, electric current and potential), influence on water properties (turbidity, pH, conductivity), secondary waste production, aerosols formation, radiological protection, etc.

In this first phase melting experiments were also undertaken using the pieces resulting from the cutting experiments.

7.1.1. Cutting by plasma-arc

Cutting parameters were first determined by Unterwassertechnikum Hannover (UWT-Hannover). Some cutting experiments have been done to prove this equipment on non-radioactive materials on a pilot-scale facility.

Afterwards the technique was successfully employed at the JEN-1 cutting facility. Pieces used had thicknesses between 3 and 60 mm and were cut 3.5 m under water. Nitrogen was used as the plasma gas and carbonic dioxide as the secondary gas. The cutting speed ranged from 10 to 120 cm/min with electric current between 200 and 500 A. Some components were partially cut on this first phase: core grid, ionization chambers support and primary cooling circuit plenum. Table II is a summary of the results obtained.

7.1.2. Mechanical cutting

Cutting experiments on aluminium were carried out on the grid support base plate of JEN-1 (25 mm in thickness). The saw blades used were 400 mm long and 3.2 mm wide. The pressure of the pneumatic advance cylinder was about 5 bar ("cutting pressure"). At these operating conditions the cutting speed ranged from 7.6 mm/min on the first cuts to 4.3 mm/min on the last ones.

TABLE I. Results of components characterization

OMPONENT	Dose Rate (mSv/h)	Gross α Emitters (Bq/g)	Gross β Emitters (Bq/g)	Gross γ Emitters (Bq/g)	Fe-55 (Bq/g)	Co-60 (Bq/g)	Cs-137 (Bq/g)	Eu-152 (Bq/g)
Core Grid	3	0.7 E00	3.0 E03	7.5 E03	2.9 E02	4.2 E03	1.4 E01	-
Grid Support	0,2	0.6 E00	8.2 E02	1.9 E03	4.3 E03	9.7 E02	9.2 E00	-
Control blade housing	30	4.2 E01	1.6 E05	1.9 E05	3.0 E04	1,2 E05	1.5 E03	1.5 E03
Ionization Chambers support	1.2	3.8 E00	1.6 E04	2.2 E04	3.1 E02	1.3 E04	1.1 E02	3.9 E02
Cooling collector	0.6	0.8 E00	1.3 E03	1.7 E03	2.5 E03	8.8 E02	3.1 E02	-
5 + 1 Fuel elements rack	0.08	0.9 E00	2.8 E01	5.5 E00	-	2.4 E00	0.6 E00	-
18 Fuel elements rack	0.08	0.6 E00	3.1 E01	7.7 E00	-	1.5 E00	2.6 E00	-
Guide tubes*	0.03	< 0.3 E00	7.1 E00	6.2 E00	-	1.1 E00	1.8 E00	-
Fuel elements NPP rack*	0.03	< 0.3 E00	5.9 E01	2.0 E01	-	7.9 E00	3.0 E00	-

* Stainless Steel

7.1.3. Melting experiments

Only aluminium materials were used in these experiments. Most of the samples came from the underwater cutting of the JEN-1 components, with very low specific activities. Hence, a number of lots consisted of pieces from the plenum, ionization chambers support and grid support. Additives, to increase slag formation and melted material fluidity and to decrease melting temperature, were occasionally incorporated with some lots. Additives, consisting of a mixture of salts (NaCl, KCl, CaF₂ in variable compositions), were added to the contaminated material at different weight percentages.

TABLE II. Results of plasma-arc cutting (1st. Phase)

PARAMETER	CORE GRID	CHAMB. SUPPORT	PLENUM
Thickness, mm	75	60	10–20
Plasma pres., psi	19 – 30	20	20
Secondary pres., psi	40	40	40
Cutting speed, cm/min	10 – 12	15	50–114
Current, A	450	200–400	225–400
Cut length, cm	24	36	1050
OTHER DATA:			
Nozzle: 3,2 mm	Plasma gas: N ₂ Distance: 5 mm	Secondary gas: CO ₂ Water depth: 3 m	

It is not easy to interpret the results. Conclusions on general lines show that decontamination factors are important only if radioactivity is present as surface contamination, but are very poor if it is mainly due to activation. Radioactivity levels in the ingots are lower than in the original material, being transferred partially to the slag. It was verified that Cs-137 and Eu-152 activities were insignificant in the ingots. Moreover, an important reduction of volume was obtained and a homogenization of radioactivity was also verified inside the ingot.

7.2. Second phase

The second phase includes cutting experiments on several aluminium and stainless steel components from the JEN-1 reactor and on stainless steel plates from the fuel element racks of a NPP.

7.2.1. Cutting by plasma-arc

Some aluminium pieces, partially cut on the first phase, were used again in this second phase, like the grid support (Al-Mg3) and primary cooling circuit plenum (Al-Mg3) as well as some stainless steel plates coming from a NPP. The following Table III is a summary of the results obtained in this phase:

TABLE III. Results of plasma-arc cutting (2nd. Phase)

PARAMETER	NPP FUEL ELEMENT RACK*	PLENUM	GRID SUPPORT
Thickness, mm	6	10	100
Plasma pres., psi	20	20	15
Secondary pres., psi	40	40	40
Cutting speed, cm/min	80	70	25
Current, A	150–200	175–200	200–450
Cut length, cm	70	150	120
OTHER DATA:			
Nozzle: 3,2 mm	Plasma gas: N ₂ Distance: 5 mm	Secondary gas: CO ₂ Water depth: 3 m	

* Stainless steel

7.2.2. Cutting by CAMC

This cutting tool has been successfully proved and handled by the five-axis positioning system and by the seven-axis slave-master robot (MHS), already mentioned.

The cutting experiments were first carried out on the 5+1 fuel element rack mock-up, (aluminium blade 3 mm thickness), and guide-tube mock-up (stainless steel), constructed to get cutting and manipulation experience with the CAMC tool.

Afterwards cutting work on radioactive aluminium was carried out on:

- A vertical aluminium beam cut from the grid support from JEN-1 (25 mm in thickness), using the five axis handling system. The results are shown in the Table IV.
- The 5+1 fuel element and the 18 fuel element aluminium racks.
- The guide-tubes for the control blades, made of stainless steel.

The results obtained in experiences b) and c) are presented in Table V.

TABLE IV. Cutting of grid support by CAMC

VOLTAGE (V d.c.)	INTENSITY (A)	JET PRESSURE (bar)	CUTTING TIME (min)	SECONDARY WASTE	
				(g)	(g/cm ²)
42	400–700	3	60	-	-
46	500–1200	4	42	270	4.5
46	300–600	4	73	290	4.8

7.2.3. Mechanical cutting

Cutting experiments on aluminium were carried out on a vertical beam of the grid support of JEN-1 (20 mm in thickness). The saw blades used were 400 mm long and 25 mm thick, 8 teeth per inch. For this configuration a 350 mm blade was useful for cutting. Summary results are shown in the Table VI.

7.2.4. Melting

Melting of stainless steel and carbon steel pieces has also been done, using non-radioactive materials. Increasing the temperature from the usual 650 °C for melting aluminium up to 1350 °C for melting steel was difficult using this equipment. Only partial melting of steel has occurred in some trials. Problems appear when the materials are partially melted, and the loading material was tried to upset. The furnace cannot keep the high temperature while upsetting. The melted material quickly cooled, solidifying inside the crucible.

TABLE V. Results of cutting by CAMC

VOLTAGE (V d.c.)	INTENSITY (A)	JET PRESSURE (bar)	CUTTING SPEED (cm/min)	CUTTING TIME (min)	CUTTING LENGTH (cm)
5 + 1 fuel element storage rack					
53	700–900	7	1.3	9	12
18 fuel element storage rack					
48	600–1000	7	1.7	16	28
48	900	7	5.0	2	10
53	900	7	0.9	98	84
53	700–800	7	3.6	10	36
Tests for control blades					
53	600–1000	8	2.5	32	80

TABLE VI. Results of mechanical cutting

CUTTING PRESSURE	CUTTING SPEED	CUTTING TIME	SECONDARY WASTE	
			(g)	(g/cm ²)
2 – 4	110	76	-	-
4 – 5	105	102	-	-
6	148	53	60	1
6	112	70	40	0,67
OTHER DATA: Feed Pressure: 6 bar Tool Pressurization: 1 bar				

8. Issues and conclusions

The main problem encountered during the thermal cutting process (plasma arc and CAMC) for aluminium is hydrosols in the cutting basin. Quick hydrosol formation limits the remote TV visibility of the cutting process. The purification system requires considerable time to remove the particles from the huge water basin volume. Also, particles must have a certain size to be retained by the filters. Therefore chemical flocculants must be added to form solid particles that can be retained by the filters. The most effective flocculant additive used for aluminium hydrosols is Na_3PO_4 , but it also requires time to work. This bad visibility or water quality usually delays the experiment starting or progressing.

The use of the above mentioned thermal technique to cut stainless steel does not present the indicated visibility problem and can be done without difficulty.

Thermal cutting techniques produce, inevitably, an appreciable quantity of aerosols, the majority of which is retained by the water in the cutting basin. To avoid the non-retained fraction disseminating into the environment, it is necessary to provide an effective gas collecting and filtering installation. In our case, the gas collecting and filtering system was very effective and our environmental monitors did not detect the presence of radioactive aerosols.

Sometimes small explosions occur in the cutting basin, due to gases being trapped as a consequence of the position of the cutting tool and the workpiece. This must be avoided. It is due mainly to water electrolysis by the electric arc. The gas measurement equipment did not detect hydrogen during the cutting process. No changes are produced in the water's physical-chemical characteristic, but a very slight increase of conductivity and pH are noted.

The results of analysis of personal dosimeters indicated that workers did not receive any significant radioactive dose during cutting and melting processes.

Environmental monitors did not record any radiation level above the background during these processes.

The cutting of a double-T shaped beam by three different cutting techniques allows one to compare them: the best behaviour was achieved by the mechanical saw cutting and the worse by the CAMC tool. The detailed conclusions of this comparative study were:

- Mechanical saw cutting presents some advantages: simple equipment, simple process, easily automated, no need for special attention from the operator, better reliable conditions, clean and good aspect of cut, narrow kerf. It produces less waste than the other techniques and no appreciable water turbidity. Thick pieces require more time, but no special geometry preparation is necessary.
- Plasma arc cutting tool is the most complicated method. It needs a water cooling system, gas feeds, and high-power electrical feed. It produces a narrow kerf (wider than mechanical cutting) with detrimental water turbidity. Not easy to position and use on thick areas to be cut.
- CAMC tool produces considerable secondary waste, a wide kerf, rude cut, with big material loss and water turbidity. Positioning is difficult and requires total dedication of a cutting operator. The cutting tool is simple, but needs secondary support systems: a water jet system and a powerful continuous high-density current source.

The mounting and operation of experimental, pilot scale, installations for underwater cutting and melting of metallic materials, close to the JEN-1 Research Reactor, has had very positive results. It has allowed the transfer of tasks to Reactor operating personnel, who have also acquired very important experience in the new decommissioning activities.

From the technical point of view, the previously mentioned events have resulted in the acquisition of know-how on dismantling and decommissioning which could reduce the technological dependency on others.

Robotic systems, like the ones used in CIEMAT to cut by the CAMC technique, are very useful from the point of view of accessibility to remote sites, reduction of dose to operators and handling simplicity.

Melting is a technique which allows an important volume reduction and a homogenization of metallic wastes. Nevertheless, its use as a decontamination technique is only effective if radioactivity is present initially as surface contamination, but it is not effective if that radioactivity is mainly due to activation.

9. Future decommissioning work in CIEMAT

CIEMAT now has a new outlook concerning decommissioning activities. An "Integral Plan to Improve CIEMAT Installations (PIMIC)" has been established and started up recently.

This is a Plan with the objective of modernizing installations and optimising the resources of CIEMAT, in order to answer the new demands of research and innovation. It consists of several types of activities such as:

- Dismantling of shut down installations:
 - JEN-1 Reactor
 - Installations for manufacturing and reprocessing nuclear fuel for research reactors
 - Installations for conditioning and storing radioactive liquid wastes
 - Other
- Modernization of buildings and facilities in operation:
 - Metallurgical hot cells
 - Test and research laboratories
 - Plant for conditioning radioactive solid wastes
 - Other
- Restoration of areas with residual contamination:
 - Buildings and soils
- Reparation of infrastructures:
 - Water distribution system
 - Other

At the end of the Plan CIEMAT will be a Research Centre integrated by:

- (i) One "nuclear island" to manage radioactive waste material, research using hot cells and other activities regulated by the Authority.

- (ii) The rest of the Centre is constituted by operating radioactive installations and conventional advanced installations or laboratories for R & D.

To carry out the activities included in the Plan, CIEMAT has an agreement with ENRESA (the national Company responsible of radioactive waste management) by which this Company is going to participate in the Plan. This collaboration includes participation of ENRESA in the dismantling of big installations and in managing radioactive waste generated in these activities.

The Plan has an industrial, not an R & D, character. Its budget amounts 6000 M Pta (36 M Euro) and its duration is from 2000 to 2006.

PIMIC implies activities with a radiological character, which are subjected to licensing processes by the Regulatory Authority (CSN). Each involved installation needs an individual license.

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Operational experience of decommissioning techniques for research reactors in the United Kingdom

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Abstract. In previous co-ordinated research projects (CRP) conducted by the IAEA no distinction was made between decommissioning activities carried out at nuclear power plants, research reactors or nuclear fuel cycle facilities. As experience was gained and technology advanced it became clear that decommissioning of research reactors had certain specific characteristics which needed a dedicated approach. It was within this context that a CRP on Decommissioning Techniques for Research Reactors was launched and conducted by the IAEA from 1997 to 2001. This paper considers the experience gained from the decommissioning of two research reactors during the course of the CRP namely: (a) the ICI Triga Mk I reactor at Billingham UK which was largely complete by the end of the research project and (b) the Argonaut 100 reactor at the Scottish Universities Research and Reactor centre at East Kilbride in Scotland which is currently in the early stages of dismantling/site operations. It is the intention of this paper with reference to the two case studies outlined above to compare the actual implementation of these works against the original proposals and identify areas that were found to be problematical and/or identify any lessons learnt.

Introduction

In total there were 36 Research reactors constructed in the UK over an approximate 30 year period, ranging from the late 1940's to the early 1970's. These reactors covered a diverse range of types and sizes as well as a number of different operators. The majority of these reactors have reached the end of their useful life and have been progressively shutdown over the past 20 years leaving only a small number that are operational today.

As with most countries attention has more recently turned to the decommissioning of these historical facilities although within the UK a range of approaches is being considered as a direct consequence of the differing technical complexities, commercial considerations and individual operator strategy. A number of facilities have opted for prompt decommissioning including the Windscale AGR, the Universities Research reactor (URR) and the Jason Reactor at Greenwich. In the case of the URR the project was successfully complete in 1996 with the delicensing of the site and the subsequent sale of the land for commercial, non-nuclear redevelopment. Alternatively certain facilities, albeit generally larger, have opted for a longer term care and maintenance approach similar to that proposed for the Magnox nuclear power plants.

Considering the range of research reactors within the UK and consequently the varied approach to decommissioning strategy BNFL were delighted to participate in the IAEA CRP on Decommissioning Techniques for Research reactors and help facilitate the exchange of practical experience gained by Member States in this field to date.

Objective

As previously highlighted the emphasis of the CRP was to concentrate on the practical experience gained by the Member States from the real-scale application of decommissioning

techniques on a diverse range of facilities. Throughout the course of the CRP programme (1997 to 2001) BNFL were engaged on fixed price contracts to decommission two research reactors within the UK namely the ICI TRIGA Reactor at ICI Billingham and the Argonaut 100 Reactor at the Scottish Universities Research Centre at East Kilbride.

Fortunately the respective contracts were at different stages of execution within the window of the CRP and allowed all aspects of the project life cycle to be captured, from initial concept design through safety justification to actual implementation works including defuelling, core dismantling and removal of all activated material in preparation for delicensing. It is the objective of this paper, with reference to the two case studies outlined above, to compare the actual implementation of these works against the original proposals and identify areas that were found to be problematical and/or identify any lessons learnt.

Case Study 1 - Decommissioning ICI Triga Mk I Reactor

1. Introduction

The ICI Triga reactor was located at Billingham, Cleveland and was used as a source of neutrons predominately for activation analysis and the commercial production of radioactive tracers.

The Mk 1 Triga was a pool reactor which operated at powers up to 250kW, using a Zirconium Hydride moderated fuel containing 8.5wt% uranium at 20% enrichment. It was commissioned in 1971 and operated until 1996. BNFL were initially awarded a contract to de-fuel the reactor and remove the activated components and ancillary equipment, leaving the reactor vessel and concrete containment intact. Subsequent to the completion of these works the contract with BNFL was extended to remove the remaining tank and concrete foundations. With the ultimate aim being to de-license the site ICI have decided to let a further third contract to BNFL to totally demolish the remaining reactor building and associated laboratories/offices.

2. Scope of work

BNFL were awarded the contract to de-fuel the reactor and remove all the activated components in February 1996. Together with the subsequent contracts the total scope of works included: -

- Development of the optimum decommissioning methodology, associated detailed design and all supporting technical justification/calculations.
- Provision of the Pre-Decommissioning Safety Reports (PDSR) and all associated safety documentation, assessments, method statements etc.
- Licensing of the fuel transfer flask
- Procurement, fabrication or provision of all the necessary equipment to support the decommissioning
- Execution of all necessary commissioning and training.
- Assistance in seeking authorization from the necessary Regulatory Authorities
- De-fuelling of the reactor

- Removal of reactor Intermediate Level Waste components
- Removal of reactor Low Level Waste components and free release waste
- Removal of the Reactor Tank and foundation concrete.
- Demolition of the reactor building, associated laboratories and offices
- Radioactive waste disposal on a fixed price basis
- Final radiological survey

3. Development of the optimum decommissioning strategy

A number of decommissioning schemes were assessed at each stage in order to determine the optimum methodology for decommissioning the ICI Triga Reactor. Design concepts were developed in parallel with the production of radiological, criticality and industrial safety assessments.

A number of factors influenced the eventual choice of preferred decommissioning methodology and this decision making process was assisted by a series of detailed HAZOP¹ Studies. These studies were undertaken with a broad representation of personnel including individuals from both BNFL and ICI to ensure that decommissioning as well as facility/operation perspectives would be applied to each problem. The main issues affecting the preferred reactor decommissioning methodology were:

- The final fuel destination was undecided at the time. A methodology was required which retained sufficient flexibility to be compatible with both UKAEA Dounreay or the USA.
- The estimated categorization of the various wastes that would be produced during the course of the decommissioning operations and their necessary condition on acceptance e.g. no free liquids.
- The physical size of the Reactor Hall was very restrictive. The limited number of viable mechanisms for the movement of transport flasks (of gross weights up to 20 tonnes) proved central to the eventual methodology. It should be noted that due to the loads involved extensive assessment of the reactor hall floor structural capability was undertaken with subsequent remedial/strengthening operations conducted in anticipation of the flasks being brought onto site.
- The Licensed Site boundary was located at a distance of only seven metres from the reactor tank. All operations needed to be conducted in a manner that minimized the dose rate to employees and the public (UK limit of 7.5S μ Sv/hr at a nuclear site boundary was mandatory).
- The Licensed Site was also located within only a short distance from local schools and shops. This imposed further restrictions in terms of the permissible potential accident scenarios. The potential for an off site occurrence, as a result of any of the decommissioning operations, had to be 'engineered out'.

¹ A Hazard and Operability study (HAZOP) involves suitably qualified and experienced people in a systematic process that is aimed at identifying both safety and operability hazards associated with the design, construction or decommissioning of any plant or process. It is a process of examination that is prompted by the use of selected guidewords and can be applied to almost any situation or set of circumstances.

- With a view to minimising the off site implications, together with the lack of suitable facilities on the site BNFL endeavoured to keep any size reduction operations to the absolute minimum.
- Considering the relatively low radiological hazard from the reactor tank and concrete foundations removal the industrial safety hazards of the proposed methodology was the primary concern. The client was very keen to minimize (or totally prevent if possible) man entry into the reactor tank and avoid all the potential hazards of confined space working.
- Also critical to the removal of the reactor tank and foundation concrete was the structural stability of the bio-shield, both during and after completion of the works. The inherent stability of the structure throughout the operation was critical to the whole concept of activated concrete removal, and therefore a detailed structural analysis needed to be performed on each potential option.

The schemes which best satisfied the above criteria, together with all the other issues resulting from the HAZOP Studies, were developed further into an engineered solution. The detailed design was co-ordinated via a dedicated design engineer who acted as a single point of contact for all such issues to ensure each interface was well defined. Having reviewed this developed scope of work, and in order to facilitate the regulatory approval process, BNFL proposed to divide the decommissioning programme into a number of discrete stages of work (see below).

4. Safety documentation

The principle document justifying the overall decommissioning approach at each stage was the Pre-Decommissioning Safety Report (PDSR). The safety case strategy was agreed as early as possible through a series of meetings conducted at ICI and importantly was developed in parallel with the preferred design solutions that resulted from the above. As with the design it was considered essential to form an integrated team comprising personnel from both BNFL and ICI. A dedicated BNFL Safety Advisor was appointed to co-ordinate the many parallel activities and provided assistance to the Project Team. Every safety case author was encouraged to attend all the HAZOP studies to ensure a thorough understanding of the project and subsequent to these meetings regular contact was maintained with the Project Team. It was insisted that draft sections of the document were issued throughout production to ensure that the correct operational information was included within the safety case as well as allowing the Project Team to brief the ICI Site Management on overall and specific issues. This document was ultimately submitted to the ICI Nuclear Safety Committee and subsequently the Nuclear Installations Inspectorate for approval.

The other supporting documentation including Method Statements, Risk Assessments and Manual Handling Assessments were produced by personnel directly involved with the proposed work on site. The Project Manager considered that this was the only manner in which a full appreciation of the activities could be gained as well as ensuring that the project personnel became fully familiar with the scope of work and decommissioning strategy.

5. Stage 1 – preparatory works

5.1. Decommissioning authorizations

Regulatory approval proved to be one of the more time consuming exercises. The major regulatory bodies involved in the decommissioning process were:

- the Nuclear Installations Inspectorate (NII), - involved with all aspects of the decommissioning programme. Agreement was required before any works could proceed
- the Environment Agency (EA), - involved with all activities involving the transportation and disposal/discharge of solid, liquid or gaseous wastes
- the Department for the Environment, Transport and Regions (DETR), - involved with issuing flask licences to allow transportation of the fuel and intermediate level waste.

The latter two issues led to a number of problems in achieving the project timescales (see Programme and Performance).

5.2. Commissioning and training

To fully test all equipment complete in-active simulation of the proposed decommissioning operations was undertaken off the ICI site utilising a flooded pit to recreate the reactor tank. This enabled load and functional testing of all manufactured equipment prior to transportation to ICI. All significant operations were trialled under the supervision of both BNFL and ICI to ensure functionality and ascertain compatibility with the reactor site. Although a minor number of changes were implemented at this stage which could have caused significant delays if encountered later on the ICI site.

Following off-site commissioning the equipment was installed within the ICI reactor hall. To facilitate this second stage of commissioning the fuel flask and necessary cranes were also delivered to the site. Full inactive commissioning was satisfactorily completed using a 'dummy' fuel element. These operations were also utilized as a training exercise and consequently all personnel identified to undertake actual operations were involved throughout commissioning to ensure a complete understanding and familiarity of the equipment and processes involved. During the course of the commissioning a number of minor improvements were implemented which simplified the eventual operations as well as contributing to significant radiological dose uptake reductions. Formal approval of the on-site commissioning was required by the NII before de-fuelling operations could commence.

6. Stage 2 – reactor de-fuelling

The reactor was defuelled utilising a cylindrical transport flask (Modular Flask) positioned directly above the reactor tank. A support frame was manufactured to provide secondary support to this flask (to ensure integrity in the event of a dropped load) in addition to a Lift and Carry Mobile Crane. This support frame also provided shielding in the form of steel and lead collimators which extended into the reactor tank water. The fuel was loaded into a purpose built fuel basket (to cater for either fuel elements and/or longer fuel followed control rods), positioned within the reactor tank next to the reactor core. When full the fuel basket was hoisted directly into the transport flask for onward shipment within a dedicated 'overpack'. The reactor had an inventory of 86 fuel elements and three control rods - hence seven transport shipments in total were required. This stage was successfully completed by December 1998 within the site programme and well below the predicted dose uptake.

A formal report of the initial shipment, comparing planned with actual operations, was required by the NII before authorization was granted to continue with the defuelling programme.

7. Stage 3 – intermediate level waste (ILW) removal

This stage constituted removal of all the ILW components of the reactor namely all the stainless steel items positioned close to the reactor core. These consisted primarily of research equipment such as the Rotary Specimen Rack (RSR) and Argon activation vessels.

A purpose built shielded container was designed and built specifically for the removal and disposal of this waste. The steel container incorporating lead shielding top and bottom included two concentric areas for waste and was positioned on the in-tank frame located next to the reactor core.

The principle issue to be considered with the eventual disposal of these items was to comply with the requirement of the waste plant operator to ensure that all consignments were devoid of any free liquid. This created a significant problem with the disconnection of the air filled RSR under water. A number of sealing methods were tested at the design phase although in order to guarantee that no liquid was present it was decided to grout the RSR in situ.

Once grouted the RSR was then allowed to cure/harden before the two connections were sheared with a hydraulic cropper. Following hydraulic cropping of the RSR connections the inner section of the container was loaded with the remaining identified components following remote size reduction with hydraulic croppers where necessary.



COMMISSIONING OF ILW CONTAINER



CROPPING OF ILW ITEMS INTO ILW CONTAINER



UNIFETCH FLASK LOADED ONTO TRANSPORTER

The support frame positioned above the reactor tank was complemented by the addition of a transfer shield. The shielded container was then removed from the reactor tank using a mobile crane. During removal, the container mated with the transfer shield to allow safe movement of the package to a Unifetch transport flask. On 27 January 1999 the waste was successfully transferred to Sellafield for interim storage pending ultimate disposal.

8. Stage 4 – removal of low level reactor waste

The remaining waste consisted primarily of the aluminium clad graphite reflector, primary and secondary cooling systems and experimental facilities such as rabbit systems etc. These

items were dismantled and placed directly into a 10m³ ISO skip for grouting and disposal at the Drigg Low Level Waste (LLW) Repository.

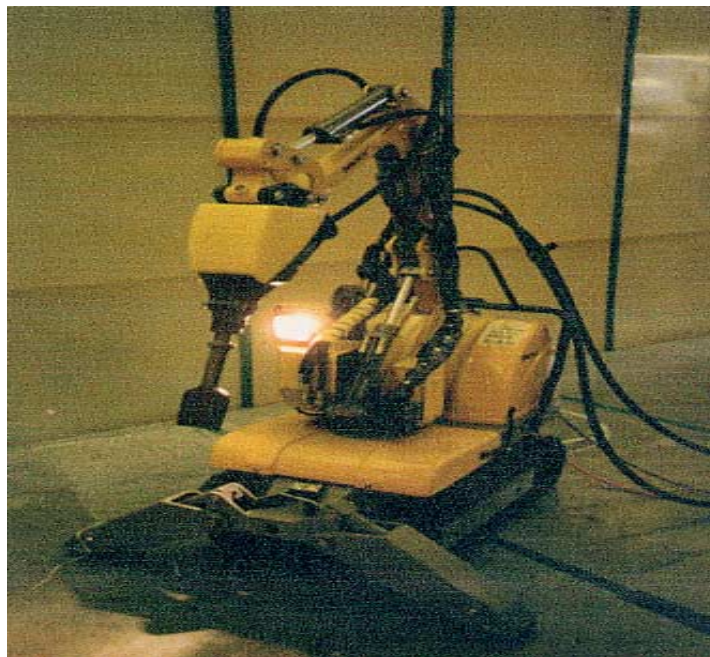
This task was successfully completed with receipt of the waste consignment at Drigg on 17 February 1999.

9. Stage 5 – removal of reactor tank and foundation concrete

As previously mentioned the client was very keen to avoid entry into the tank because of the potential risks of confined space working and therefore strongly favoured the remote solution offered by using a Brokk Minicut. This mini-excavator could be operated, via the use of cameras, from outside the tank and could undertake all the necessary tasks involved i.e. concrete breaking, cutting metal components and waste removal

In preparation for the task a modular containment, modified lifting equipment and a dedicated ventilation system was installed over the drained reactor tank. Once this had been commissioned both the tank and surrounding concrete was subjected to extensive sampling and analysis to determine the profile for active concrete breakout.

The first operation to be performed was the removal of the aluminium tank walls up to 2m from the tank base. Specially selected cutting discs were acquired and adapted to be compatible with the Brokk Minicut. When all of the tank wall aluminium was successfully removed concrete breakout commenced. – see below. The depth of breakout was routinely checked using a standard laser distance-measuring device.

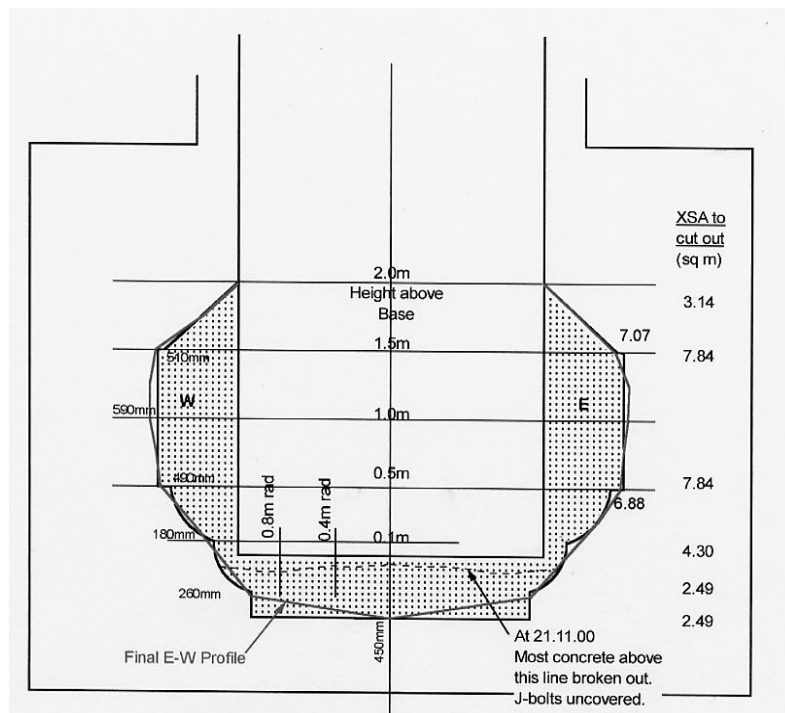


BROKK MINICUT WITH CONCRETE BREAKER

9.1. Base removal

When it was deemed that sufficient concrete had been removed from the walls, the aluminium base was removed with the grinding attachment. Once the tank base was removed the concrete was broken out as before.

The profile of the broken out concrete is shown on the drawing below. The final profile matches the original proposed breakout profile, except at the outer extremities of the base.



FINAL CONCRETE BREAKOUT PROFILE AGAINST PROPOSED PROFILE

10. Programme and performance

An interesting feature of the programme shows that the original timescales allocated for the application/granting of the necessary regulatory approvals were consistently underestimated. This was particularly evident in the case flask licensing which needed to be completed by the owners of the fuel flask. Not only was this task started late due to the prolonged decision making process on the final destination of the fuel and consequently which flask to use, the duration of licensing itself exceeded the original estimate by over 50%.

Apart from the above the two primary issues were the unforeseen requirement for reinforcement of the reactor hall floor and ICI's major uncertainty with the ultimate fuel destination. There were two final possible destinations available (Dounreay and the USA) and in order to retain flexibility ICI wished the development of a scheme compatible with both. This necessitated the review of the proposed scheme, required additional design effort/time and the nomination of a new fuel flask to ensure that this compatibility was maintained. When the decision to dispatch to the US was made BNFL agreed to arrange interim storage of the fuel as an extension to contract pending the eventual onward transfer. This change of scope further delayed the original project programme whilst an application for regulatory approval was submitted to support this new requirement for interim fuel storage.

Once the above issues had been resolved which significantly lengthened the preparatory works programme the actual site works were completed in a shorter timescale than originally anticipated i.e. 20 weeks as opposed to 26 weeks. This trouble free implementation is deemed to be a direct consequence of the considerable time and effort that was dedicated during the design development, safety case preparation and commissioning.

With respect to the removal of the aluminium tank and concrete the initial works proceeded as programmed although there were a few minor difficulties with the aluminium tank. The major

issue was that actual concrete breakout rate was considerably lower than rate derived from the inactive trials. On investigation this was attributed to the specific nature of the reactor concrete.

11. Safety

11.1. Radiological dose uptake & health physics data

As can be seen from below there was a significant difference between the predicted and actual total project dose uptakes of 19.98 mSv and 1.57 mSv respectively. One explanation is that defuelling was considerably delayed hence the fuel had experienced far greater cooling. Other factors included over estimating working times and using worst case activation levels. However this dose uptake still compares very favourably with all previous decommissioning projects of this complexity and emphasizes the importance of developing the design in parallel with the safety cases, rigorous training/commissioning and strict supervision. With respect to the tank and foundation concrete dose measurements taken after removal of the reactor demonstrated that there was no significant radiological hazard. This was confirmed by actual dose measurement during operations.

11.2. Potential accident scenarios

Following the extensive HAZOP studies three potential accident scenarios could not be discounted and needed to be further addressed in order to justify the safety of the project as a whole.

ACTUAL VS PREDICTED RADIOLOGICAL DOSE UPTAKE

TASK	ACTUAL DOSE /mSv	PREDICTED DOSE ² /mSv
Stage 1		
Site Preparation and Inactive Commissioning	0.054	1.15
Stage 2		
Active Commissioning and Reactor Defuelling	0.239	5.65
Stage 3		
ILW Removal Operations	0.058	6.21
Stage 4		
LLW Removal Operations	1.219	6.97
Total Dose Uptake	1.570	19.98

² Predicted Dose based on Pre-Decommissioning Safety Report - Dose Assessment - April 1997 - Based on decommissioning operations commencing March 1998

These comprized:

- Possible dropping and damage to the ILW container shielding on transfer to the Unifetch Flask.
- Possible damage to the graphite reflector leading to the possibility of Carbon 14 contamination
- Accidental damage to the core by a falling fuel flask (weight c. 10 Te).

These scenarios each required extensive modelling and assessment to justify the ultimate safety of the proposed decommissioning approach. Each assessment resulted in key amendments to the detailed to ensure that all the necessary criteria were met.

11.3. Conventional safety

The main conventional safety hazards were identified as water hazards, electrocution, moving machinery, work at height, possible confined spaces and lifting and handling operations. In conjunction with ICI, BNFL ensured that all individuals involved with the project were suitably qualified and experienced for each specific task. In addition for each task detailed method statements, incorporating risk assessments were generated by the individuals involved and submitted to ICI for approval before commencement of any works.

11.4. Safety performance

No injuries or accidents occurred to any member of the decommissioning team including BNFL, ICI and contract personnel. This was directly attributable to the planning which took place, particularly during the design and commissioning of the equipment, combined with the training regimes all employees underwent.

12. Project management & resourcing

The development of an integrated project management team was considered to be one of the most important contributing factors to the overall success of the project. The core project team was very small (3 BNFL and only 1 full time ICI) and hence all support from both organizations was very focused and well co-ordinated. It was considered imperative that the same team was maintained throughout the life of the project, from design through to implementation, and that each individual was involved with every aspect of the project development. The small, dedicated, team also promoted an in depth understanding of the project strategy from conception, provided an appreciation of how/why the proposed decommissioning methodology had been developed and a detailed knowledge of the proposed implementation. In addition, because the team incorporated both BNFL and ICI personnel, with specialist support resources where necessary every task was considered from a range of perspectives and hence ensured that each task could be successfully implemented on the ICI site.

A close working relationship was developed within the integrated management team as both parties had the fundamental priority of completing the project safely. This helped move BNFL/ICI away from a contractually based approach and allowed a greater flexibility with respect to the deliverables and accountabilities of both organizations. This working relationship built on trust and honesty was mutually beneficial and promoted the single commitment towards safe and effective project delivery. Such a relationship was considered essential to the success of this project.

13. Current status and further work

The ultimate aim of ICI, the client, is to de-license the site. During discussions with the NII, it was decided that the only sensible way to satisfy all the necessary criteria would be to totally demolish the facility. Consequently a third contract has recently been awarded to BNFL to demolish the reactor building, associated laboratories and offices.

The current indicative programme is as follows:

Preparation of safety documentation — June–October 2001

Soft strip building — January 2002–June 2002

Demolish building August 2002

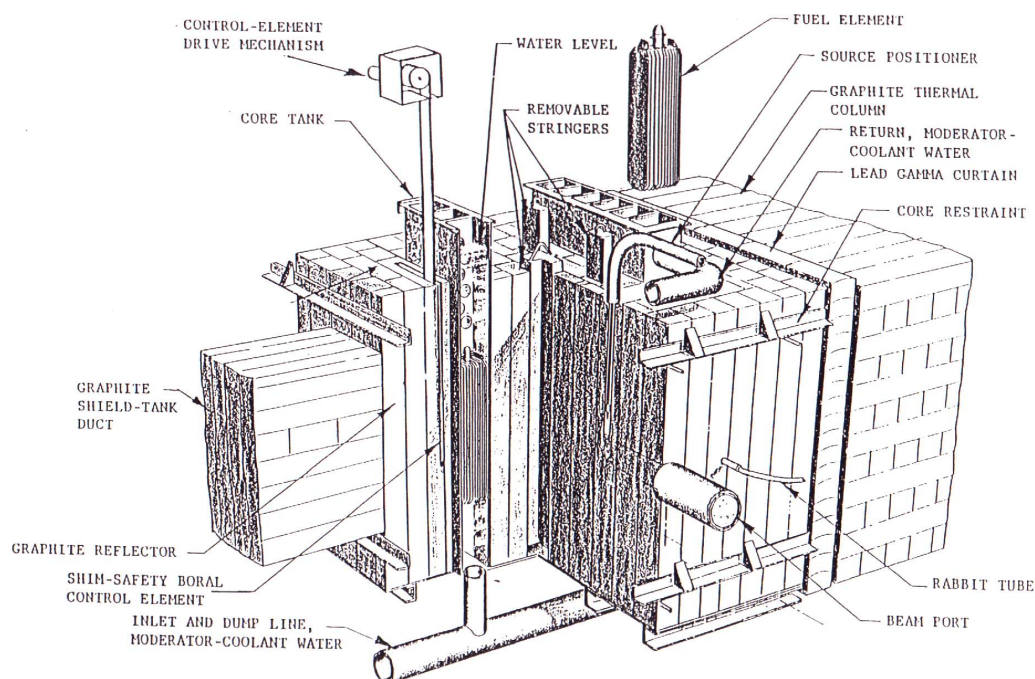
Final surveys application to de-license March 2003

Case Study 2 - Decommissioning of the Scottish Universities Research Reactor (SURR)

1. Introduction

The Scottish Universities Research Reactor (SURR) is owned by the universities of Glasgow, Edinburgh and Strathclyde and is situated at East Kilbride, Glasgow. The UTR Type Reactor first went critical in 1963 with a maximum power of 100KW although it was up-rated to 300KW in the early 1970s. In the early 1990s a decision was made to decommission the reactor and demolish the reactor hall with the ultimate objective of de-licensing the site. The reactor was defuelled in January 1996 and a large Co-60 irradiation source was removed in early 1999 (by BNFL under a previous contract). SURR conducted a competitive tender evaluation to complete the remaining decommissioning and awarded a contract to BNFL to undertake these works in May 1999.

The reactor core consists of a carbon stack in which are embedded two parallel core tanks, which previously contained the fuel around which water circulated (see diagram below). The water was returned to a dump tank adjacent to the reactor. The whole reactor is contained within a reinforced concrete bio-shield.



SECTION THROUGH THE REACTOR CORE

2. Scope of work and proposed methodology

Principally the required scope of the project and the proposed methodology can be defined in following main areas of work:

- Safety justifications and supporting documentation/evidence
- Preparatory work including installation of flexible containment system suspended from roof structure of reactor hall with dedicated ventilation/filtration system.
- Remove free release concrete from outer faces of reactor monolith using Brokk 330 (to minimize the risk of cross contamination from later LLW removal).
- Remove shield blocks and graphite using Brokk and gantry crane.
- Remove ILW using Brokk, Mini Cut and specially designed cutting tool.
- Break out remaining LLW concrete and foundations.
- Strip out active ventilation and drain lines.
- Demolish, undertake comprehensive radiological survey and de-licence

3. Development of the optimum decommissioning strategy

From the radiological data supplied by the client tender stage one of the initial tasks of the project was to undertake extensive sampling and analysis to characterize all resulting wastes. One of the more significant requirements was to identify the boundary between free release waste (FRW) concrete and low level waste (LLW) within the concrete monolith. The method utilized was to remove/analyse a series concrete cores from the outside face of the bio-shield. Extrapolation of these results was further supported by a number of physical samples obtained from the inside faces. Based on these estimates, the waste was split into three categories: -

- (a) Free release waste, estimated to be the outer 1-1.5m of the concrete bio-shield.
- (b) Low level waste (LLW), the remainder of the concrete bio-shield, the graphite from the core and thermal columns, and miscellaneous aluminium pipe-work within the concrete.
- (c) Intermediate level waste (ILW), principally all the steel components with in the reactor core.

The methodology was based around a remotely operated vehicle, in this case a Brokk 330, with various tools used to remove the reactor components. The strategy adopted was to remove the FRW before activated material to avoid cross contamination. A primary containment was required to prevent spread of contamination during active operations.

4. Safety documentation

The Pre Decommissioning Safety Report was again the principal safety document and was supported by the necessary design justification reports. In the particular case of the ventilation design the design engineer attended the HAZOP meetings and provided the safety case writer with relevant technical information.

To ensure continuity of work a separate sub- safety case was produced to justify erection of the containment structure in advance of the main PDSR.

The project team also produced a series of method statements and risk assessments to support all the installation activities and commissioning documents were produced to control testing

of the installed plant and equipment. All of this documentation was presented to the SURRC Nuclear Safety Committee and subsequently the NII for approval.

5. REGULATORY AUTHORIZATIONS

As explained in Case Study 1 the principle UK Regulators were the NII. The necessary NII approvals to proceed were obtained as required although this is largely attributed to early and regular contact between the NII and the client. The client again included BNFL staff in all Regulatory discussions which ensured an essential level of mutual understanding.

Because of the geographic location waste authorizations were undertaken by the Scottish Environmental Protection Agency (SEPA). Unfortunately these became a critical programme issue with their expected granting to take in excess of twice the programmed duration of 15 months. Another aspect of the waste authorization was the requirement to measure aerial releases of tritium, even though the Radiological Risk Assessment predicted the worst case scenario to be within acceptable limits. An additional tritium measuring system needed to be developed to sample from the building ventilation system. As Free Release Waste (FRW) is exempted from authorization this phase of the work was able to proceed – see below.

6. Decommissioning work to date

All of the Free Release Concrete has now been removed from the reactor bio-shield, using the Brokk with concrete breaker and clamshell bucket for waste removal (see below)



BROKK BREAKING OUT FREE RELEASE CONCRETE

When sufficient concrete was broken out, samples were taken both from the broken concrete and the remaining concrete on the reactor. Once confirmed as FRW the concrete was loaded into a hopper, which was wheeled out of the containment and tipped into a FRW skip located outside the reactor building.

The original core sampling provided the basis for the detailed approach of each removal campaign. Concrete was generally removed to a depth of 0.5m in the first pass before the bio-shield was again subjected to confirmatory sampling/analysis. Work proceeded on that basis until all free release concrete had been removed.



REACTOR AFTER REMOVAL OF FREE RELEASE CONCRETE

During the design stages it was anticipated that substantial quantities of dust would be generated from the removal methodology. Although minimization of dust generation had been incorporated into the original design continual improvements and minor modification of the methodology throughout the operation delivered a 70% reduction in the quantity of dust.

7. Outstanding/completion issues

When waste authorizations are issued the project will resume with removal of the reactor core, active concrete and removal of ancillary equipment prior to building demolition. Currently trials of the ILW size reduction machine are progressing. The machine has again been adapted to fit on the Brokk and will size reduce the core base plate for disposal in an existing and approved ILW liner.

A separate safety case has been prepared for the building demolition and is currently being reviewed by the client.

8. Lessons learnt

The principle lessons learnt can largely be attributed to either of the two case studies and hence are presented below as an independent section:

- A clear and concise understanding of the overall decommissioning problem and definition of the exact deliverables is necessary before any development of a strategy is undertaken.
- Review all the parameters that have the ability to affect the optimum design scheme e.g. requirements of waste plant operators, specific site restrictions due to location, physical access, etc.
- Generally research reactors were designed to be operated and not to be decommissioned. Consequently unexpected demands may be made on the reactor and surrounding areas e.g. the excessive loadings which necessitated reinforcement of the ICI reactor hall floor.
- Wherever possible physically confirm all information taken from existing drawings of the facility. Never assume that drawings accurately reflect the actual status of plant.
- Develop safety documentation in parallel to the engineering solution as they are inherently linked.
- Endeavour to provide flexibility within the detailed design to overcome perceived uncertainties.
- Undertake extensive risk assessments from a hazard and operability view point to assist the design process. Involve representatives from a broad cross section of disciplines, including existing operational reactor staff.
- The initial stages of the project are key to the eventual success. Ensure sufficient time is allocated for the development of the optimum strategy and the detailed design. Incorrect decisions at this critical stage of project development could have a very onerous effect on both safety and costs at a later date.
- Allow adequate time within the programme for tasks outside the direct control of the project team e.g. the submission of applications for regulatory approval.
- Appoint a dedicated and integrated core management team comprising individuals from both a decommissioning and operational background. Ensure continuity of the team is maintained from project conception, through design and implementation, to completion.
- The integrated management team need to co-ordinate all aspects of the project to gain a complete knowledge base of the project. Although specialist support resources may be required, tasks (particularly the development of the necessary site documentation) should be undertaken directly by this team wherever possible providing that they are suitably qualified and experienced.
- Undertake comprehensive commissioning and training. Where possible utilize inactive mock-ups, away from the nuclear site to demonstrate and improve the operation of the equipment. Once delivered to site it is recommended that inactive trials of all equipment are then conducted to confirm compatibility/functionality before undertaking active operations. This also allows personnel to familiarize themselves with the proposed tasks as well as provide the opportunity to include any identified improvements within the methodology.

- Continual improvements should be sought throughout the decommissioning operations, particularly those of longer duration e.g. dust management at the SURRC. Consider combining methodologies to get the best elements of both i.e. for ICI concrete removal a number of schemes were considered at the design stage however experience suggests that a combination of techniques may have been more effective.
- Activated concrete contains substantial quantities of tritium. If potential for aerial release exists the regulator may require this release to be quantified and methods of measurement will be required.
- A variety of standard tools can easily be adapted to suit a variety of decommissioning and remote tasks saving expensive development of specialized tools and equipment.

Integration of improved decontamination and characterization technologies in the decommissioning of the CP-5 research reactor, United States of America

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Abstract. The aging of research reactors worldwide has resulted in a heightened awareness in the international decommissioning community of the timeliness to review and address the needs of research reactor operators in planning for and eventually performing the decommissioning of these types of facilities. Many reactors already undergoing decommissioning can be used as test beds for evaluating enhanced or new/innovative technologies for decommissioning; it is possible that new techniques could be made available for future research reactor-decommissioning projects. Potentially, the new technologies will result in: reduced radiation doses to the work force, larger safety margins in performing decommissioning and cost and schedule savings to the decommissioners in performing the decommissioning of these facilities. Testing of these enhanced technologies for decontamination, dismantling, characterization, remote operations and worker protection are critical to furthering advancements in the technical specialty of decommissioning. Furthermore, regulatory acceptance and routine utilization for future research reactor decommissioning will be assured by testing and developing these technologies in realistically contaminated environments prior to their use in actual research reactor decommissioning. The decommissioning of the CP-5 Research Reactor located at the ANL-East Site has been completed. In this paper we present results of work performed at Argonne National Laboratory (ANL) in the development, testing and deployment of innovative and/or enhanced technologies for the decommissioning of research reactors. In addition, details are provided on other related U.S. D&D activities, which may be useful to the international research reactor D&D community.

1. Objectives

By using reactors already undergoing decommissioning as test beds for evaluating enhanced or new/innovative technologies for decommissioning, it is possible that new techniques will be available for future research reactor decommissioning projects which will result in a number of favorable conditions. These include: reduced radiation doses to the work force, larger safety margins in performing decommissioning and cost and schedule savings in performing the decommissioning of these facilities. Testing of these enhanced technologies for decontamination, dismantling, characterization, remote operations and worker protection are critical to furthering advancements in the technical specialty of decommissioning. Regulatory acceptance and routine utilization for future research reactor decommissioning will be assured by testing and using these technologies in realistically contaminated environments.

Over the past 2–3 years, various technologies were evaluated for their potential future routine use in decommissioning. These evaluations occurred during the conduct of the CP-5 Decommissioning Project. Selected technologies were demonstrated at the facility and their effectiveness benchmarked against other baseline technologies for those same technical topical areas. The focus of the technology demonstrations was on the areas of the reactor, fuel pool, hot cell and the rod storage area.

2. Introduction

Worldwide, there are a large number of research reactors which have been shutdown and are awaiting decommissioning. [1] There has been an increased emphasis on the use of improved decommissioning technologies to address these shutdown facilities. Largely as a result of an out growth from the Large Scale Demonstration Projects (LSDP) on the CP-5 research reactor at ANL-East, several other LSDP's (see Table 1) have been commenced (and in some cases completed) at other U. S. DOE sites [2–3] which also focused on D&D technologies and reactor D&D projects. In some instances, other LSDPs actually re-deployed and further fine-tuned technologies previously demonstrated at CP-5. In other instances different technologies were deployed than were used at CP-5.

Table 1. U.S. Department of Energy large scale D&D technology testing projects

Site	Facility Type
Argonne National Laboratory	Research Reactor
Fernald Site	Uranium Production
Hanford Site	Production Reactor
Idaho National Engineering & Environmental Laboratory	Fuel Pools
Los Alamos National Laboratory	Glovebox Laboratories
Mound Site	Tritium Facility
Savannah River Site	Fuel Fabrication Facility

It is also important to point out that as a part of our program to attempt to optimize the use of improved D&D technologies, it is also critical to optimize exchange of lessons learned in the broad area of D&D (including deactivation). As a part of this activity, the ANL-East D&D Program has initiated a D&D training course available to the international D&D technical community, which now allows for the information exchange process to occur through the facility-decommissioning phase in the entire life cycle of a nuclear facility.

3. CP-5 Research reactor site history/description

A major component in the U.S. DOE Decommissioning Program was the dismantlement of the CP-5 Research Reactor detailed further in this section [4].

The Chicago Pile No. 5 (CP-5) Research Reactor Facility, situated on approximately three acres in the southwestern section of the Laboratory, was the principal nuclear reactor used from 1954 to 1979 for

the production of neutrons for scientific research. During its lifetime, the reactor generated in excess of 5.4×10^8 thermal kilowatt-hours and was used to irradiate more than 27,000 samples for research purposes. In September 1979, the reactor was shutdown for the final time, and in 1980 the facility was placed into dry lay-up pending funding for decommissioning.

During decontamination and decommissioning (D&D) of this facility [5] all radioactive components, equipment, and structures associated with CP-5 were disassembled and removed. In late fiscal year (FY) 1999, the Laboratory determined that there is no reuse for the CP-5 building structure. Therefore, the goal of this project was to leave the building in a final physical end state that allows the structure to be demolished in the future when appropriate funding becomes available. Figures 1–7 show some of the final site activities to release the facility for eventual demolition.



FIG. 1. CP5 research reactor facility.



FIG. 2. BROKK Excavator removing activated concrete.



FIG. 3. Concrete section of fuel pool being removed for disposal.

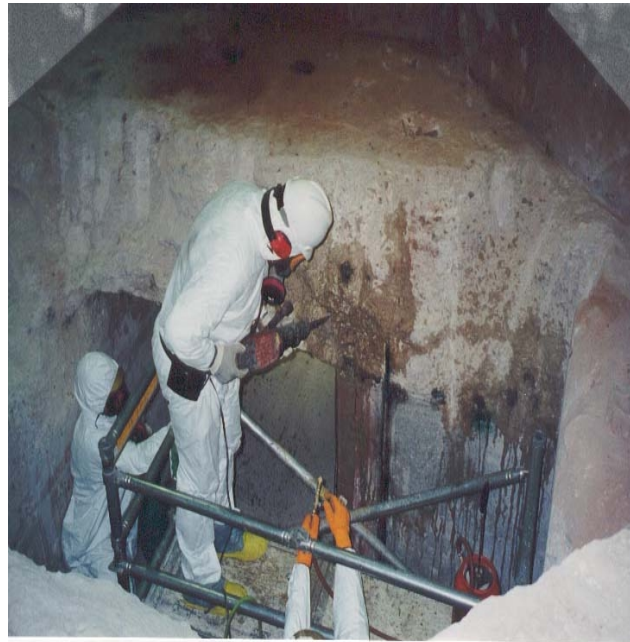


Figure 4. Concrete removal from reactor pedestal area.



FIG. 5. Polar crane removal from containment.

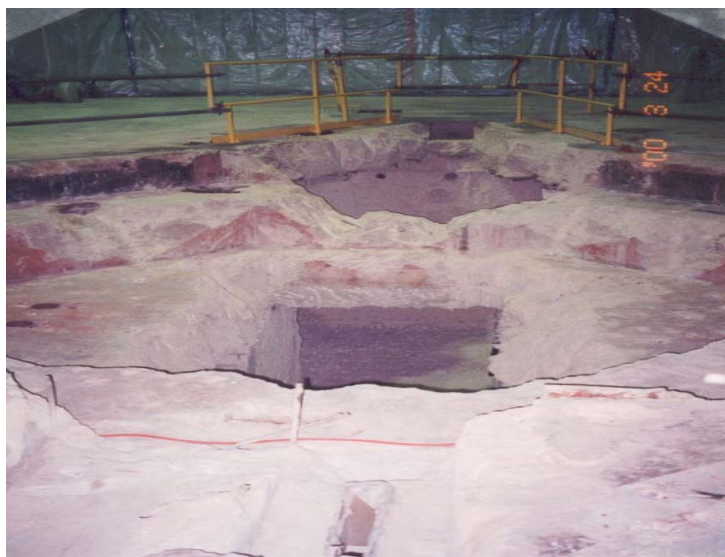


FIG. 6. Excavated area around reactor.



FIG. 7. Vaporshere structure was free released and re-used by Plant Services.

At the conclusion of the Independent Verification Contractor (IVC) survey, the CP-5 facility was released as an “Industrial Use Area,” indicating that residual elevated activity remains in inaccessible areas. In accessible areas, however, exposure levels are at background; Radiation Work Permits (RWPs), radiation monitors, and other radiological controls are not needed to enter.

All accessible areas of CP-5 facility were decontaminated to radiological background levels. The reactor vessel and internals; contaminated piping systems; and miscellaneous tanks, pumps, and associated equipment were decontaminated and/or packaged as radioactive waste. Process equipment and associated plumbing, ductwork, drain lines, etc. were dismantled; the reactor vessel internals were size-reduced in place; the reactor weight plugs and experimental equipment stored in the spent fuel pool were removed; all radioactive waste was packaged, and shipment and disposal properly documented; and a thorough survey of the facility after the removal of activated and contaminated materials was performed. CP-5 major systems removed, e.g., packaging of experimental components, fuel racks and reactor fuel weight plugs (located in the fuel pool); clean out of the hot cell and storage tubes; and removal of fuel pool, hot cell liner, and reactor internals, was performed by in house personnel. The removal of the bioshield concrete, rod storage tubes, hot cell, High Efficiency Particulate Air (HEPA) ventilation system, fuel pool transfer slot, and the final decontamination of remaining equipment, floors, walls, and ceilings were performed by a subcontractor under the management of the D&D Program technical staff. A final radiological survey was performed by the subcontractor with an IVC survey being performed by the Oak Ridge Institute for Science and Education. The total project duration was 97 months. Decommissioning activities were initiated in June 1991, and the final report was issued in July 2000. The total exposure to project personnel was 0.12 man-Sv (11.5 person-rem), with no individual exceeding the CP-5 D&D Project limit of 10 mSv (1 rem). The total cost of the CP-5 D&D Project, including labor, management, and waste disposal, was \$29.5M. A total of nearly 1.1×10^6 kg (2.4×10^6 pounds) of radioactive waste, including liners, was packaged for off-site disposal at a low level radioactive waste repository, with a total activity of 4.9×10^{13} Bq (1310 Curies). Additionally, over 219,000 kilograms (482,000 pounds) of mixed waste, including liners, with a total activity of 6.11×10^{12} Bq (165 Curies) was packaged for disposal. In July 2000, the CP-5 Facility was formally decommissioned and transferred to the site landlord.

4. Technology deployments

The need for new or enhanced decommissioning technologies is derived from both experiences in the field and from an assessment of the costs and safety implications of the various operational steps in D&D. For example, in cases where there is a large cost associated with a specific phase of decommissioning, payoffs from improvements could be significant and, therefore, worth the investment in the development of the new technologies.

At ANL and the other DOE national laboratories, work is in progress along with support from universities and other private industries, that when taken together involves developing a better as well as a fuller set of technologies — some novel and innovative, other adaptations of existing technologies — to better suit the needs for future D&D tasks. The area in which technology development work is being carried out at ANL is listed in Table 2. Specific references [6–11] are provided where further details are available on these technologies.

The ANL technology development work ranges from basic research (for example, in developing a fundamental understanding of the chemical bonds on contaminated surfaces — which can lead to the development of methods to “surgically” break the bonds with minimal energy expenditure and secondary waste production), to purely applications development (for example, design and development of interface hardware and software to apply robotics technology to specific applications). On an intermediate level is the development of a system to utilize ANL developed chelating agents for chemical decontamination of piping. Present data suggest that this method produces high decontamination factors with a considerable reduction in secondary waste generation compared to current methods. Another example is the demonstration of the use of laser systems with fiber optic beam delivery systems for use in decontamination and cutting operations.

Prior to its acceptance by a commercial D&D operation/entity, it is necessary to demonstrate a laboratory-developed technology under near prototypic conditions. A number of technologies that show promise in a laboratory setting may prove to be unusable in the contaminated radioactive environment typical of a D&D operation. A number of the research and test reactors at ANL and other DOE sites have reached the end of their operating lives and are being used as tests beds for the demonstration of enhanced D&D technologies. In conjunction with the ongoing D&D of the CP-5 reactor, a program of demonstrations of the D&D technologies was undertaken under the sponsorship of the U.S. Department of Energy’s Environmental Restoration and Technology Development Programs. The program was managed by the Strategic Alliance for Environmental Restoration comprised of ANL, two nuclear utilities, two large engineering companies and a major university [12].

A formal technology selection process was defined and utilized for the selection of appropriate technologies for demonstration at CP-5 as well as at the other LSDP sites. The selected technology demonstrators were required to develop detailed test plans and safety assessments, and receive final approval for testing contingent upon approvals of the test plan or safety assessments as well as fulfill several institutional and financial considerations.

The technologies selected could be classified into four distinct categories:

- Facility Characterization
- Robotics/Dismantlement
- Facility and Equipment Decontamination
- Worker Protection/ Containment

For the CP-5 D&D project, in total, 86 innovative technologies were evaluated as part of this process with 41 of these accepted for demonstration. For a number of reasons, 18 of the selected technologies were unable to go through the entire demonstration process and twenty-three technologies were actually demonstrated. The demonstration process consisted of the significant size of the demonstration (e.g., square foot area to be decontaminated; length of piping to be examined, etc.) within which the innovative technology was operated. Data was taken on various technical, safety and cost parameters and these were compared against corresponding data for baseline technologies. The improvements (or lack thereof) in performance were tabulated. The database so compiled is clearly going to be extremely valuable for D&D planners in making their decisions on the appropriate technologies for their D&D operations.

Of the 23 technologies demonstrated, 9 were characterization technologies, 8 were decontamination technologies, 4 were dismantlement technologies, and 2 were worker health and safety technologies. Several of these presented significant benefits to the D&D operations while others were only marginal beneficial. Final reports on the technologies and data have been widely available to the D&D community. Detailed evaluation results have been prepared, and distributed and more will be in the near future. A project final report has also been prepared and distributed. These evaluations will assist planners of future D&D operations in selecting appropriate technologies for the work.

Table 2. Technologies Under Development at Argonne National Laboratory

- Chemical Decontamination
- Robotics Applications
- Laser Decontamination
- Cutting Technologies
- Sodium-Cooled Reactor D&D Technologies

5. Technology deployment results

In this section, details are provided on selected technologies which might be of greatest interest to others planning for decommissioning of research reactors. Many of these technologies were deployed and evaluated at CP-5 and others were deployed at other DOE LSDPs.

Surface Contamination Monitor and Survey Information Management System (SCM/SIMS) [13]

The SCM/SIMS is designed to perform both alpha and beta radiation surveys and to then document the measured data. A motor driven cart houses the position-sensitive gas proportional counter with a variable width of detection ranging from 0.5–5.0 meters (20–196 inches). The SIMS component of SCM/SIMS is a series of software programs, which processes and analyzes the collected data into either a standardized or a customized data report.

For our test area, this technology was from 2 to 28 times faster than the manual survey baseline techniques. The fact that the system automatically generated the data reports with minimal operator intervention was the greatest benefit.

GammaCamTM radiation imaging system [14]

The GammaCamTM system is designed to provide remote two-dimensional information on both the positive and relative strengths of gamma ray radiation fields. The source of the gamma field can be from a few to several hundred feet away from the observer. The system consists of a portable sensor head and a portable computer for control. The sensor head contains both gamma ray and visual imaging systems. The benefit of this technology is primarily in reduced worker doses and it also provides a two dimensional color image of gamma radiation fields on a corresponding black and white photo image.

This approach, when compared to the baseline approach, was found to be 3 times faster.

In situ object counting system (ISOCS) [15]

The ISOCS is a portable, in situ Germanium based spectroscopy system specifically designed to provide information on types and amounts of radioactive material. The system consists of a Germanium detector/portable cryostat; a cart to hold the unit; lead shielding and collimators, a portable spectroscopy analyzer; along with a portable personal computer and the in situ calibration software.

When this system was deployed and tested at CP-5, it was easy to use and provided reasonable agreement between the baseline data and that obtained using ISOCS. The most significant benefit to using ISOCS was that it provided real time, non-intrusive assay information without the typical delays often encountered in off-site sample analysis. The cost of ISOCS was about 70% of the baseline cost of off-site analysis.

Pipe ExplorerTM [16]

The Pipe ExplorerTM System is a characterization technique/method for transporting a variety of tools into piping or ducting. The system uses a pneumatically operated airtight tubular membrane as a protective envelope for towing radiation detectors and video cameras into pipe. The membrane envelope is pressurized and then provides a safe, clean conduit for the sensors to travel through. This system can be used up to 61 meters (200 feet) away from the base unit and in piping/ducting ranging from 5 to 102 centimeters (2 to 40 inches) in diameter. Deployments have been completed at both commercial NPP and DOE sites. The protective membrane feature allows workers to avoid handling potentially contaminated materials and also prevents contamination of the equipment in the membrane. In addition, since the membrane slowly rolls, the contamination does not move with the membrane.

Centrifugal shot blast system [17]

This system consists of a shot blast unit which propels steel shot at a high rate of speed in order to abrade concrete and concrete coated surface. Spent shot is collected by a vacuum system and reused until fully spent. Fully spent shot is vacuumed into a HEPA filtered dust collection system 208 liters drums (55 gallon drums). The application of this technology at CP-5 was for the removal of contaminated paint from a concrete floor area of about 74.3 square meters (800 square feet). The removal rate using this technique was about 29 square meters (310 square feet) per hour – nearly 50% over the baseline production rate. This unit is self propelled and reduced operator fatigue; the use of the vacuum dust/debris collection system also assisted in reducing airborne dust concentrations.

Dual Arm Work Platform [18]

The Dual Arm Work Platform (DAWP) is a robotic system used to work in radioactive and other hazardous work areas where exposure levels preclude or limit human intervention. The DAWP consists of platform base, two schilling Titan III six degrees-of-freedom hydraulically driven manipulators, a remote viewing system, a lighting system, a tool control system and a tether that supplies hydraulics, power and control signals to drive the DAWP functions.

The DAWP was used at the CP-5 facility as a work platform for dismantlement of the reactor vessel (RV) bioshield complex. When compared to the baseline technology for RV/bioshield complex removal, the DAWP resulted in about 50% savings in cost over the baseline approach.

3M EmporeTM membrane separation technology [19]

This membrane separation technology provides a method to cleanup radioactively contaminated water.

The technique employed consists of enmeshing surface-active particles in a web-like matrix, which has been formed into a membrane. The membrane takes the form of a cartridge filter for use in standard commercial cartridge filter housing. At high flow rates, this technology is capable of removing beta and gamma particles to below detectable limits. The technology was used at CP-5 to remove soluble Cs-137 and Co-60 from the fuel pool water. Empore was used to process over 17,000 liters (4500 gallons) of pool water at a rate of 1.9 liters (0.5 gallons) per minute. The initial concentrations of Cs-137 and Co-60 were 20 mBq/l (0.6 pCi/l) and 7 mBq/l (0.2 pCi/l) respectively; after filtering the levels of both dropped to below 0.7 mBq/l (0.02 pCi/l).

Benefits of the technology included: generation of less secondary waste, cartridge sorbents can be selected for specific contaminant, and the sorbent material was more efficient than an ion-exchange treatment option. The experience at CP-5 resulted in the ability to process 2460 liters (650 gallons) of water/day at a cost of about 50% of the baseline approach of evaporation.

Lead paint analyzer [20]

This analyzer is a hand held battery operated unit, which uses x ray fluorescence to identify not only lead in paint but up to 25 other elements. The unit has an 8-hour rechargeable battery, results as available in 30 seconds and it is able to store up to 3,000 data points/measurements. Although the initial capital outlay is rather large at \$25,000, the cost savings and schedule savings allow for (in one case) full recovery of the capital cost by using it on only 20 samples as compared to submitting the samples to an outside laboratory for analysis.

Remote Underwater Characterization System (RUCS) [21]

RUCS was used on the INEEL LSDP to identify objects on a fuel pool floor, which were not previously visible to operators due to water clarity problems in the pool. The RUCS unit was an “off the shelf” model with a color camera and GM tube radiation detector fit to it. An auto depth control unit and other operation and control systems were located on a central control panel. Due to its “quick” availability and lack of the need to perform significant design changes, an estimated 40% cost saving resulted from using this technology over other options/solutions.

Diamond Concrete Shaver [22]

This technology was developed in Europe and was deployed at the DOE-Hanford Site at one of the former production reactors, the “C” Reactor. It consists of a 25 centimeters (10-inch) wide diamond impregnated shaving drum propelled by an electric motor. Depths of decontamination can range from 0.013–1.3 centimeters (0.005–0.5 inches). When compared to the baseline technique of a pneumatic scabbler, a 67% cost savings and a five fold increase in removal rates was realized. This service is now available through a nuclear decontamination service provider in the United States.

En-Vac Robotic Wall Scabbler [23]

The En-Vac Robotic Wall Scabbler technology was deployed at the INEEL LSDP. It consists of a remote controlled scabbler unit with individually motor controlled wheels, which can work on horizontal and/or vertical surfaces. Vertical surface use is facilitated by the use of a high vacuum suction system. A five-fold increase in production rate was noted over the use of handheld scabbling equipment. Other noted benefits were lower waste volumes and lower worker radiation doses.

Mobile integrated temporary utility system (MITUS) [24]

Many older facilities may have poor “as built” drawings on facility configuration. At the Hanford Site and many other DOE and other older sites, this is a major D&D concern. To alleviate this concern, Hanford deployed the MITUS technology to serve as a mobile temporary power supply and communication system for D&D project locations. This unit is trailer mounted and can support up to 20 sub-units within a working area. Full electrical service support (120, 240 and 480 volt) plus alarm system and other notification systems is unit integrated into MITUS. This unit can eliminate safety concerns of project staff in the conduct of D&D of these older facilities.

Automatic locking scaffold [25]

Often D&D involves working at heights over areas where there is no means to gain access to equipment/systems requiring removal. It is often necessary to use temporary scaffolding in these areas to perform the required D&D activities. Standard scaffolding is often labor intensive and cumbersome but with this automatic locking scaffold system, the equipment is much easier to set up, faster to set up, and has fewer parts to track and use in the assembly process. The design minimizes use of the tools to erect the scaffolding and even facilitates worker “tie off” locations for safety.

Personal Ice Cooling System [26]

The Personal Ice Cooling System (PICS) is an outgrowth of a cooling suit developed and used by Canadian pilots to cool aircraft flight personnel in the Gulf War. Ice water is circulated through tubing in a liquid cooling garment worn by the workers. Ice water in a bottle/insulated bag is worn on the hip or back. These weigh only about 5.4 kilograms (12 pounds) and when used increased productivity by over 200%. Cost savings were nearly 50% when compared to the PPE, which would have been worn otherwise. These suits are widely used throughout the US DOE Complex on D&D projects and other routine operations.

Oxy gasoline cutting torch [27]

This is a technology that actually poses much less risk than a first review of the technology might make a user believe. The technology has been in use for many years-its application in D&D is a new approach to an old problem – size reduction of large carbon steel components. The system uses a 9.5

liters (2.5-gallons) gasoline fuel tank, flow shutoff valve, hoses and cutting torch. Cutting of thick carbon steel 1.3–11.4 centimeters (0.5–4.5 inches) thick is possible allowing a 10 centimeters (4 inches) thick carbon steel plate to be cut at a rate of 127 centimeters (50 inches) per hour compared to the oxy-acetylene cutting rate of 25.4 centimeters (10 inches) per hour.

BROKK [28]

The BROKK device is actually a mobile demolition robot that is controlled by either a tether or a radio remote control. The key attributes of BROKK that greatly assist in D&D work are reduced worker fatigue, reduced likelihood of heat stress and lower radiation exposures. Multiple end effectors can be deployed on it including: scabblers, hydraulic shears, hammer, grapple and bucket. Use of the BROKK resulted in over a 90% cost savings and significant schedule savings on several deployments to date. This technology has now become the baseline technique for concrete removal.

Vec Loader HEPA-Vacuum System [29]

The Vec Loader HEPA Vacuum System was demonstrated at the DOE-Fernald Site for use in dust and debris collection and asbestos removal. The unit has a diesel-powered engine that produces a suction to remove 48.1 cubic meter (1700 cubic feet) of material per minute at 38 centimeters (15 inches) of mercury vacuum. The HEPA filter traps and contains liquid, slurry and solid waste constituents. The unit has a 152 meters (500-foot) vacuum hose, which can be routed to the work area.

Soft – Sided Waste Containers [30]

This technology was demonstrated at a D&D project site at the INEEL. Historically, low level radiation waste at many sites has been packaged and disposal of in metal drums or boxes. The soft sided waste container is constructed of an outer shell and inner liner of polypropylene approved as a U.S. Department of Transportation IP-1 package. When compared to a conventional metal waste container, the soft-sided waste container has a capacity of 7.36 cubic meter (260 cubic feet) compared to 2.72 cubic meters (96 cubic feet) for a standard metal box. In addition these containers are 70% cheaper to procure than the standard metal waste containers.

Copper Wire Recycle System [31]

This technology was also deployed and demonstrated at the INEEL site. The NUKEM system consists of a conveyor, a pre-shredder and grinder. Insulation is removed from various feed sources of copper cable with the clean copper being recovered for recycle/reuse. Based on the deployment at INEEL, about 5080 kilograms (10,000 pounds) of material was processed in an 8-hour work period. Instead of disposing of this material as waste or undertaking a very labor-intensive recycling effort, a more economical method is now available to allow the recovery and recycle of the materials.

6. CP-5 Decommissioning project-operational lessons learned

The decommissioning of a research reactor has associated with it many unforeseen difficulties that may be unique to a particular facility. However, the solution to these problems may be applicable to situations encountered during other similar D&D projects. The following are problems encountered, the solution to those problems, and the resulting lessons learned. There were four near-miss safety incidents that occurred during project execution. Details are provided in the following paragraphs on each incident.

Dropping of E-Wing Crane Block in Pool: Two workers were removing pool liner sheets from the bottom of pool. The crane block was lowered near the bottom of the pool. The crane cable came off of the drum and struck one worker on the shoulder. The worker was examined by the Medical Department, who reported the worker had suffered a minor bruise. The operator knew the block was unable to reach the bottom of the pool; standard operating procedure was to leave three wraps of cable on the drum at all times. The crane was inspected and load tested. A line indicating the lower limit for block travel was painted on the pool wall visible to the crane operator and the crew was retrained.

Lessons Learned: Although this incident resulted from inattention to detail by a trained crane operator, the incident indicated the need to be alert at all times to potential problems associated with crane operation. Facility managers and crane operators should be aware of any limitations on crane operations. Facility signage and operator aids should be used to help ensure that crane operators are alert to and aware of such limitations.

Failure of Nylon Lifting Strap: The second incident involved the failure of a nylon-lifting strap while attempting to lift a beam gate from the biological shield. It was subsequently determined that the sling failure was caused by the beam gate being attached to the biological shield and containing sharp edges, which apparently resulted in cutting the nylon sling. Several factors contributed to the incident, including 1) lack of as-built component drawings depicting attachments, 2) an extremely high radiation field that did not allow for a through visual inspection to ensure a free and clear lift, and 3) existence of sharp edges.

Several lessons learned and corrective actions were implemented to address this incident, including: 1) prior to the subsequent removal of the beam gate, a thorough assessment of the engineering drawing was performed to determine if there were any additional steel members or pipes associated with the gate; 2) prior to rigging, equipment was inspected for sharp edges and softeners were put into place; 3) retraining of all crane operators; 4) all lifting and rigging operations required the use of the load cell attached to the crane hook; and 5) dual verification of all crane lifts was required.

Top Cap Separated from Drill Bit: The third incident occurred during the removal of a core drill bit in the rod storage area. During the lift, the top cap unexpectedly separated from the bit and was projected upward. The apparent cause was a poor design of the drill bit cap. In accordance with corrective actions from the previous event, a load cell should have been used during the lift. Due to operator error, a load cell was not used at the time of the incident. Use of a load cell could have possibly prevented the cap from separating by indicating the load tension. A Stop Work Order was initiated by the Project Manager, and a work stand-down took place until the incident was fully investigated and corrective measures implemented. All project personnel were counseled on the importance of safety and the contractor's safety requirements and expectations. Also, as a result of this incident, additional personnel were added to the project, including a certified crane operator and a health and safety professional.

Lock-Out/Tag-Out Not Verified: The fourth incident involved an employee taking samples in a retention tank without verifying a Lock-Out/Tag-Out (LO/TO) was in place. This was a procedural violation and as a result, a Stop-Work Order was issued by the project manager.

During this work stoppage, the contractor performed a casual factor analysis to determine the cause of the incident, and to identify and implement lessons learned and take appropriate corrective actions. The contractor developed and implemented additional management procedures and conducted training for all project personnel and also conducted an independent readiness assessment to ensure that all requirements were being implemented in a safe manner. Subsequently, the project manager performed a readi-

ness assessment that did not identify any items to prevent resumption of work. In addition to the new management procedure, all workers received additional, augmented training in the following areas:

- Safe work practices
- Job safety analysis
- Hazardous communications
- Radiation safety
- Lockout/Tagout
- Lead awareness
- Health and Safety Plan
- Lifting and rigging
- Confined space
- Job/task analysis
- Pre-job briefs
- Stop-work authority

Thorough knowledge of historic operations is a key factor of quality characterization, especially at experimental facilities: The lack of detailed documentation for the CP-5 reactor operations significantly impeded the characterization and subsequent D&D activities performed during the project. The lack of detailed drawings resulted in conservative assumptions in the cost estimate, including the waste volumes, activity durations, and personnel radiation exposures.

Continuous safety presence at the job site: Either the ANL Project Manager, Project Engineer, Field Engineer, or Safety Engineer were within the facility the majority of the time to observe ongoing activities. All project personnel not in the shell were in the office area just outside containment. During potentially dangerous activities (i.e., operation of the BROKK), an assigned safety observer was used to stop work in case of emergency. These safety precautions resulted in no significant injuries to personnel.

Bins designed for high-density waste for metal and bioshield concrete were used: According to the original work plan, 1.93 cubic meters (68 cubic feet) bins with a 3628 kilograms (8,000 pounds) weight capacity were to be used to load concrete and metal. The weight limit would be reached long before the volume was filled, resulting in large amounts of void space. A decision was made to use 1.36 cubic meters (48 cubic feet) bins with a 4536 kilograms (10,000 pounds) weight capacity. The result was an increase in material placed in each box, and a decrease in the amount of void space.

Compactible trash was used to fill voids in all bins. Compactible trash (i.e., used protective clothing, wipe-down rags, etc.) was used to fill in the void spaces that remained in waste bins.

“Niche” contractors can be very productive: The use of contractors to perform certain activities reduced both the overall cost and time expenditure of the project. Contractor personnel may have expertise in needed areas that are otherwise unavailable and are economically competitive.

Continuous review of lessons learned: D&D projects on the scale of the CP-5 D&D project will experience unforeseen complications and difficulties. Additionally, certain procedures may be significantly improved in both time and cost efficiencies once both are refined. By continuously reviewing the lessons learned, mistakes and/or inefficiencies can be corrected earlier, resulting in a smoother, less costly, and more productive project.

7. Applications to research reactors worldwide

As stated earlier, a large number of research facilities will need to be decommissioned in the future. The experience and expertise necessary to undertake this task is often in short supply in many of the countries where the reactors are located. The experiences gained by all operators of research and test reactors in decommissioning operations, technology development and demonstrations should be utilized in assisting the future organizations responsible for planning and execution of decommissioning of their facilities. This will ensure optimization of the decommissioning process.

8. Conclusion

A variety of decommissioning technologies have been and still are being deployed at various research reactor and other D&D sites across the country and world. This final report summarizes the ANL CP-5 D&D experience (both operationally and technologically) and provides additional detail on technologies from the CP-5 research reactor decommissioning project [32–34].

9. Future work

The CP-5 Research Reactor has now been fully decommissioned. However, the completion of the project does not represent the end of the decommissioning activities here at our site. There are still numerous facilities requiring decommissioning currently identified on the ANL-East Site including: the Juggernaut Research Reactor located in the Building 335 High Bay, a Hot Cell facility (Building 301), several Zero Power Reactor facilities (Building 315) and a set of 10 liquid waste storage tanks situated in Building 310. Preparations to initiate decommissioning or actual dismantling work is currently underway at each of these facilities.

Decommissioning of the 60-Inch Cyclotron Facility in Building 211 was just recently completed and the final report issued for that work. [35]

Integration of improved and enhanced technologies into the conduct of the above decommissioning activities is an on-going activity. Really the integration and the use of these technologies across the entire DOE Complex and even other non-DOE facilities as is an on-going activity of numerous groups within different organizations. The US Department of Energy Large Scale Demonstration Projects have stimulated the urgent need for and use of improved technologies in performing future nuclear facility decommissioning activities. This is essential if we are ever to be able to complete the decommissioning of all surplus facilities within a tight budgetary situation and within the ever tightening compressed time frame that others desire.

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