Engineering

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Retrieval and Segregation Task 2.2 Final Report

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Engineering Report

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EUROPEAN COMMISSION
Community Research

CARBOWASTE

Treatment and Disposal of Irradiated Graphite and Other Carbonaceous Waste

Grant Agreement Number: FP7-211333

Report (D2.2.1)

WP 2 RETRIEVAL AND SEGREGATION

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Gateshead, United Kingdom

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Document title
WP 2 Retrieval and Segregation Task 2.2 Final Report
Executive summary

Deliverable D2.2.1 is the output of a study covering retrieval and segregation of legacy waste, Task 2.2. The study is continuing with diagnostic and modelling, Task 2.3. This report considers retrieval and segregation in the context of decommissioning commercial reactors in less than 25 years or up to 75 years after shutdown. Retrieval and segregation of i-graphite stored in an unconditioned state is also reviewed and the analysis is expected to be of use to research and test reactors. A team of specialists from UK (Doosan Babcock Energy, NNL, AMEC), France (EdF), Lithuania (LEI, PI), Spain (ENRESA) and Romania (INR) with support from other CARBOWASTE beneficiaries contributed to the report.

WP1 provided input to the study including specific country information, graphite data, reactor systems, past retrieval experience, a route map and upper level evaluation of retrieval criteria. A Data Pack, technical report for T 2.1.1, was produced and used as input to a Work Package 2 workshop. (see SINTER).

There are many requirements to achieve optimum retrieval and segregation. Firstly, the pre-requisites of good nuclear practice, the route map and the project management process. Secondly, the definition of the preliminary waste route and its integration with the integrated waste management process. Thirdly, a staged approach to selecting the methods and tools. A generic retrieval and segregation process is defined that can be broadly applied to any reactor system. The parameters and inputs needed to select each of the process steps have been developed. It was established that in-air or underwater retrieval and segregation are likely to be the dominant working environments and criteria are presented for evaluating these and a trial Multi-Criteria Decision Analysis is reported. The report considers planned methods for decommissioning Magnox, UNGG and RBMK reactors and identifies challenges and alternative approaches. The result of the analysis has led to a generic upper level work breakdown structure and a number of checklists to aid solution engineering.

To proceed to a solution the preliminary waste route must be defined that fits the integrated waste management strategy for the total i-graphite management and specific reactor design and condition knowledge should be obtained. Then, to fully manage risk it is important to control contamination, protect the work force against radiation and achieve good visibility of the operational workface. These parameters affect the choice of working environment (in-air or underwater); but there is no single solution and tooling selection must be on a case by case basis. Tooling suggestions are noted based on prior experience.

Going forward modelling will continue in Task 2.3 to cover some of the uncertainties identified so far.

Revisions						
Rev.	Date	Short description	Author	Internal Review	Task Leader	WP Leader
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02	10/02/10	Draft v2	MJG	PF	PN	

1.0 Contents

2.0 Preamble

The legacy waste aspects of Work Package 2 comprise:

> Task 2.1: Radiological, reactor design and graphite mechanical integrity considerations. A technical report CARBOWASTE-0904-T-2.1.1 has been produced (SINTER).
>
> Task 2.2: Examination of retrieval and segregation options. A report of the workshop has been produced (SINTER) and this report is the deliverable for this task.
>
> Task 2.3: Diagnostic and Modelling Activities. This work is in progress.

This report is the first of two formal reports to be produced in Work Package 2 covering the retrieval and segregation of legacy i-graphite. Several other documents have been produced and stored on SINTER that are relevant to this study.

The next report will cover Diagnostics and Modelling activity and is due to be issued round about month 37. At that point this report will be re-addressed and will reference the key results of diagnostics and modelling report. It will not be changed in substance but will be checked for consistency and for reference to new tools developed. It is intended that the two reports will need to be read together and neither will supersede the other.

Consequently, there is an opportunity from time to time for Work Package 2 beneficiaries to supplement the information in this report either by the production of attachments that can be appended or by minor changes to the text here in.

Any such comments or changes or recommendations should be sent to Doosan Babcock Energy Ltd at Gateshead for consideration.

Introduction

3.1 Background to the Report

The CARBOWASTE Project comprises 6 work packages:

WP1. Integrated Waste Management Approach
WP2. Retrieval and Segregation
WP3. Characterisation and Modelling
WP4. Treatment and Purification
WP5. Recycling and New Products
WP6. Disposal Behaviour of Graphite and Carbonaceous Waste

This report is concerned with Work Package 2 of the CARBOWASTE project, Retrieval and Segregation, and is specific to legacy waste. It does not consider those aspects of Work Package 2 concerned with the retrieval and segregation of irradiated graphite in future High Temperature Reactors (HTGR). Any further reference to Work Package 2 in this report should be read in the context of legacy waste and Tasks 2.1 to 2.3.

IMPORTANT NOTE.
The report generally refers only to the retrieval and segregation of i-graphite from within reactor core containments (often but not always a pressure vessel). Some of the information will be equally relevant to i-graphite that has been removed from a reactor prior to decommissioning and is stored in an unconditioned / unpackaged state. A report on the retrieval of such wastes at Vandellos, Spain has been included as the benchmark for retrieval of stored i-graphite. Some of this information is of more general value.

The report is deliverable D-2.2.1 Report Retrieval and Segregation prepared in response to the CARBOWASTE Description of Work [1] .

The objective of the CARBOWASTE Work Package 2 tasks is to develop a toolbox compartment of best practices and management approaches integrated within the CARBOWASTE toolbox. The latter covers the retrieval, treatment and disposal of irradiated graphite (i-graphite). A number of countries are considering the strategies and options for the identification, retrieval, treatment and final disposal of i-graphite and the first step in the process will be to retrieve it (or possibly not) from the reactor containment. It is essential that this retrieval process is compatible with the down stream waste management activities as well as total decommissioning policies and strategies. The CARBOWASTE project takes account of these national activities and assimilates their considerations against appropriate end points.

In this context it is possible that the i-graphite material will need to be segregated either into separate categories to take account of variation in activity, form and endpoint, or from other materials that may be associated with it within the reactor containment or during its retrieval.

Within the European Union, total retrieval and/or segregation of i-graphite materials has been successfully completed on a number of facilities, which for reactor core graphite GLEEP and Windscale AGR are notable, and for operational fuel element debris, Vandellos I vaults can be considered a reference case. Outside the EU, most notably in the USA at Fort St Vrain and in Russia at Leningrad NPP retrieval experience of i-graphite is noted.

This report is primarily, though not exclusively, concerned with total removal of graphite from commercial scale graphite moderated reactors that dominate the EU's i-graphite inventory and for which solutions are the most urgent. These are the Magnox, UNGG and RBMK plant operated in UK, Italy, France, Spain and Lithuania. In the future these will be joined by the AGR operated in the UK. Whilst these commercial scale reactors dominate the requirement, the report will be of relevance to the retrieval of i-graphite from research and test reactors and for graphite wastes already removed during operation and stored in an unconditioned / unpackaged state.

Due to the number of local variations in reactor design, operation and regulation, it is not reasonable to develop specific solutions for every reactor type so this report focuses on developing a generic approach to retrieval and segregation that can then be tailored to specific retrieval projects by a variety of users working to policies and strategies that will vary amongst the EU States. Reactors that have operated normally have not been reviewed as these have case specific challenges.

At least two environmental modes of dismantling (in air and in water) have been used or developed in some detail at the time of writing. In air was the mode at GLEEP and Windscale AGR; water is the accepted medium that will be used when irradiated graphite is removed from Bugey in France. Whilst water is considered acceptable for many of the UNGG range of reactors, underwater dismantling may not be optimum in all cases. So this report examines the key issues relevant to choosing the most appropriate environment and processes.

The literature on the management of irradiated graphite waste is extensive and prior to start of the CARBOWASTE project the most recent document of international importance that comprehensively covers this subject is IAEA TECDOC 1521[2]. Other knowledge on managing legacy i-graphite (as well as operational i-graphite) has been reported at various conferences and training courses, some very specific to i-graphite. The most notable during the timescale of the CARBOWASTE project have included EPRI workshops at Kendal [3] , Lyon [4] and Hamburg [5] , an IAEA conference in Manchester [6] and a University of Manchester Graphite Training Course encouraged by the CARBOWASTE project itself. A key feature of all of these has been the involvement of the organisations and many individuals working on CARBOWASTE. However, very little has been reported at these events about the specific process of retrieval and segregation of legacy graphite. If it has, then it has been very application specific e.g. GLEEP or Windscale AGR or has been on some particular element of concern; an example is the work done by EPRI to demonstrate how difficult it is to create a graphite explosion or fire. No toolbox guide exists to direct the engineer through the process to follow to derive an i-graphite retrieval and segregation process that interfaces with and is optimised within the integrated i-graphite waste management solution.

Such a solution will be the sum of a generic approach that can be followed by any member state or utility and project specific activity, which could vary from place to place and organisation to organisation. This Retrieval and Segregation report pulls together in to one place information on the retrieval and segregation of graphite and identifies processes (tools) to assist users of this document to map and plan an irradiated graphite retrieval project. In support of this goal, Work Package 2 draws upon the first major review task of the CARBOWASTE project, which was a Review Report [7] undertaken by Work Package 1. This report is predominantly a review of the status of i-graphite management in most countries in the world that have i-graphite but it also covers other issues such as the graphite quality and waste management.

3.2 Participants

The main participants leading to the production of this report are summarised in Table 1. This identifies those who were employed by beneficiaries of Work Package 2, and those who attended the Work Package 2 Workshop, which was the key formative event associated with Work Package 2. These organisations and participants were comprehensive in terms of background and interest in irradiated graphite and have either taken part in group discussion or provided reports and information leading to the knowledge base in this report. A number of individuals brought interest from other Work Packages to the project or were invited because they were at an early stage in their formation and training. These have been identified in the table 1.

The background and experience of the organisations employing these individuals may be found in the CARBOWASTE Description of Work [1].

The organisations and personal involved in Work Package 2 represented experience and interest in the countries to which they belong, activities in research and education, development, nuclear plant operation and project management and implementation contracting and decommissioning liability management. Those organisations identified with * also provided information to the review report of Work Package 1 [7] in the form of status report relevant to i-graphite management for their own and other countries.

Table 1 – List of Participants

NAME	AFFILIATION	WORK PACKAGE 2	WP2 WORKSHOP
Anthony Banford	NNL UK * & WP1	√ and WP1	√
Francis Berenger	EdF * & WP3	√	√
Dominic Chadwick (University of Manchester)	PhD Student		√
Daniela Diacanu	INR Romania	√	
Grigorijus Duskesas	FI Lithuania	√	√
Harry Eccles	NNL UK * & WP1	√	
Paul Fisher	Doosan Babcock UK *	√	√ and Facilitator
Francisco Madrid García	ENRESA Spain *	√	
Jon Goodwin	Bradtec UK WP5		√
Michael Grave	Doosan Babcock UK *	Technical Lead	Convenor
Joanne Hill	AMEC UK	√	
Lionel Maciver (Doosan Babcock)	Graduate Trainee	√	√
Jon Montgomerie	AMEC UK	√	√
Povilas Poskas	LEI Lithuania *	√	√
David Ross	NNL UK		Lead Facilitator
Csaba Roth	INR Romania	√	
Werner von Lensa	FZJ Germany * WP4		√

3.3 Method

The activities to derive the CARBOWASTE Retrieval and Segregation Toolbox Compartment for legacy i-graphite wastes are as follows. WP2 has proceeded in 3 steps following on from initial activity in Work Package 1.

- *Task 2.1, Radiological, reactor design and graphite mechanical integrity considerations.* This resulted in a technical report T-2.1.1 Data Pack [8] that was a precursor for the following tasks, both strategically and in the provision of historical information relevant to retrieval and segregation.
- *Task 2.2, Examination of retrieval and segregation options.* This step was the main activity of the Work Package 2 team and the focus of development was a workshop, where the output of Task 2.1 was thoroughly reviewed, and further developed in to this report deliverable D-2.2.1.
- *Task 2.3, Diagnostic and Modelling activities,* Arising from the work up to task 2.2 was the identification of areas where modelling would be beneficial to support the i-graphite retrieval and segregation decision process. This step has proceeded to further develop priority areas. It will be fully reported later. The output of Task 3 will be deliverable D-2.3.1, Diagnostic and Modelling Report.
- *Completion of Task 2.2* Upon completion of Task 2.3 the report in Task 2.2 will be updated to take account of modelling developments, this report is designated T-2.2.2.

The integration of each step of a route map in to an integrated waste management strategy, which is compatible with ecological, economic and socio-political requirements, is a key feature of the CARBOWASTE project so, before Work Package 2 commenced, information was required from Work Package 1 that addresses this integrated approach. The Work Package 1 information comprised, firstly, the route map for total i-graphite management from retrieval to ultimate disposal (including possible treatment and reuse on the way). Within this route map, Step 2 covered retrieval and segregation itself and examined the pre-requisites and potential end states. The total integrated route map is illustrated in figure 4 and Step 2 of it in figure 5. Secondly, Work Package 1 produced a review report which included amongst other information, reference to prior activity of relevance to retrieval and segregation and this information is referenced in this report as required and also in a Data Pack[8].

An important link between Work Package 1 and Work Package 2 is Task 1.6 that considers retrieval and segregation in the context of the integrated waste management solution, specifically examining the upper level criteria and upper level retrieval options that might be used to meet downstream processing and disposal practice. Task 1.6 is reported separately [14].

Whilst retrieval and segregation of i-graphite needs to be integrated with the waste management process it will, in most cases, be linked with other reactor decommissioning activities and such links are noted within the report. The impact of radiological and core integrity effects on the retrieval process over a range of time horizons is considered.

The CARBOWASTE route map [9] and critical decision analysis developed by Work Package 1 has identified two key stages, viz. the current disposition of i-graphite, i.e. in the redundant reactor, and its final disposal. A toolbox approach is used by CARBOWASTE to assist users to take forward and develop optimum solutions. This report defines a generic process that fits into this framework, referred to as the “retrieval and segregation tool box compartment” and considers the issues that affect the design process.

The Work Package 2 workshop focused primarily on Magnox, RBMK and UNGG reactor designs and decommissioning policies that might arise in the UK, France and Lithuania. Consequently, as well as being able to examine the differences in approach due to three contrasting generic reactor designs, the issues of decommissioning early (i.e. 25 year or less) or late (i.e. about 75 years) and in-air and underwater were compared and contrasted. The consideration of these issues is a major feature of this report as also is the need to clearly understand the preliminary waste route, the need to be able to access the graphite and the need to deal with the condition of i-graphite. The approaches in the workshop were also challenged by considering taking account of very specific approaches and challenges to be found in dealing with legacy HTR plant in Germany.

4.0 Scope

4.1 Preamble

This report covers the retrieval and associated segregation activities of i-graphite from the core region of a range of nuclear reactor configurations following reactor shutdown for final decommissioning. For completeness, where relevant, reference is made to operational wastes.

The dominant designs of reactor analysed are Magnox (operated in UK, Italy and Japan, RBMK (operated in Lithuania, Russia and Ukraine), and the commercial UNGG designs (operated in France and Spain).

This report has, however, attempted to be generic so that its content may be applicable, as far as is possible, to other commercial reactor systems of which the dominant is AGR (operated in the UK), to research reactors (several designs operated in a number of countries including the German HTR legacy plant) and also to operational wastes that have already been removed from reactor containments but remain unconditioned / packaged.

The report references lessons learned, where available, from the existing experience of removing core and other graphite from Windscale AGR, GLEEP, Fort St Vrain and Russian reactors, operational waste graphite from silos at Vandellos as well as conceptual design and front end engineering undertaken in France (for Bugey), in Japan and in Germany. In developing from these experiences and lessons it draws upon published or otherwise volunteered information resulting from a pre-decommissioning considerations in UK, France, Lithuania, Spain and Germany, which includes consideration of the methodology for access to the graphite, tooling for retrieving of it and the impact of alternative time scenarios – rationalised in CARBOWASTE to less than 25, 50 and 75 years after reactor shutdown.

The report identifies generic and common features of removal of graphite bearing in mind the route map developed in Work Package 1. Whilst it is recognised that in many cases removal of graphite is integrated into the total reactor dismantling activity, the report is limited to issues concerning access to the graphite, handling graphite and its removal from the reactor containment but key interfaces with other decommissioning activities are identified. Operational wastes may have been retrieved from some reactor designs during the operational lifetime; these retrieval operations that are usually part of the fuel route have not been considered but the report does reference some work that has related to retrieval of such waste from silos or vaults. Where Work Package 2 team members had experience of sampling i-graphite from operational reactors use of any relevant information was taken in to account if considered of use to legacy i-graphite retrieval. This report does not, however, cover sampling activities in detail as they are considered to be well established and have been carried out on various reactor designs for a number of years.

4.2 Prerequisites, terminal points and assumptions

The compilers of this report worked in the context of prerequisites that would be expected to be in place, terminal points that might be attained and in accordance with a number of assumptions. The user of this report needs to be familiar with these.

Further it will be necessary to consider that constraints, controls or project states (not necessarily specific to the irradiated graphite retrieval and segregation challenge) are in place before giving specific consideration to the best solution for retrieval and segregation of the i-graphite itself. Of priority amongst these will be the understanding of terminal points

The i-graphite retrieval task may well be covered by international standards, national policy and regulation, owner rules, integrated decommissioning programmes and reactor common design issues. Such issues will exist for any decommissioning project and it is not within the scope of this report to examine these but only to note the need to take them in to account in developing any retrieval and segregation solution. A number of exclusions and bounding conditions have been identified below that were recognised but not further engineered by the Work Package 2 team.

4.2.1 Fundamental Management Prerequisites and Good Practice

a) Steps will be taken to ensure that funds are available to enable reactor decommissioning and/or the associated graphite retrieval and segregation operations to proceed

b) A safety culture will be adopted focused on eliminating harm to or misuse of persons, plant, materials, energy and environment.

c) Recognised Quality Systems appropriate for decommissioning operations and waste management will be in place.

d) These Quality Systems will apply to all aspects of the process, planning and design, safety case approval, implementation and waste management

e) Processes will be designed and operated to minimise the volume of radioactive and other wastes.

f) Processes used for retrieval and segregation will be those that are proven by prior use or, where this is not the case, are fully tested, validated and demonstrated to be safe, practicable, reliable, economic and socially acceptable.

g) The tools, materials and skills necessary to support the processes in d) and e) will be available from the supply chain and resource pool. Where this is not the case it will be possible to demonstrate that these requirements are within economic and practicable reach.

h) Personnel used in the i-graphite retrieval and segregation process at all stages will be suitably qualified and experienced (SQEP), will be adequately trained and able to demonstrate appropriate professional qualities.

4.2.2 Policy and Regulatory Approach

A generic approach adopted in this report cannot take full account of differences that might be encountered in different member states, so the user of this report should take account of the following when developing specific solutions.

a) Policy and Regulatory Approach may not be specific to graphite and may not need to be specific to graphite in all areas. It is more likely to be specific to graphite in the area of Waste Management than in the area of Health & Safety.

b) Any extant EU regulatory directives will be assumed to apply if such exist and also where relevant any IAEA guidance.

c) Specific national policies and laws relating to the retrieval and segregation activity or any other associated programme of activity of which it is part, integrated or impacted shall be taken in to account.

d) The specific requirements with respect to Health and Safety, Waste Management, Qualification of processes or individuals shall be taken into account for the country where the graphite retrieval and segregation will take place.

e) Where no prior experience of irradiated graphite management exists in a specific country then it is recommended that regulators are approached and involved in the project at an early date and where lessons can be learned from the development and application of policies and regulations in other countries, such information is obtained wherever possible. Reference is also made to the Work Package 1 Review Report, where review of the regulatory systems in several countries may be found. As far as the Work Package Review team is aware, only France has defined specific law with respect to the disposal of irradiated graphite. The selection of the final repository is in progress at the time of writing of this report. UK has, however, obtained approval for packaging of irradiate graphite from WAGR and for the treatment of very low level graphite from GLEEP.

f) Emergency plans are expected to be in place for any external impact from a nuclear power plant being decommissioned. It is not envisaged that such events will happen as a result of graphite retrieval or segregation.

4.2.3 Exclusions

This document does not consider whether 25, 50, 75 year or indeed any other timescale is most appropriate for removing the i-graphite. These periods are considered to be end-points or constraints around which a retrieval and segregation process must be designed.

The retrieval and segregation tool box compartment does not consider the total plant decommissioning process. Where it is the case that i-graphite is removed at one or more stages of reactor decommissioning sequences, it is taken for granted that the decommissioning organisation will consider the interaction of the various i-graphite and non-graphite stages. This integration is project specific and outside the scope of this report. Other decommissioning steps are noted to be constraints on the i-graphite process and in turn other steps may be constrained by the i-graphite requirements.

The report has not considered the development of technologies with time. The findings are based on the current knowledge and understanding of the participating beneficiaries.

The report does not take account of the fact that regulatory standards may change with time. The findings are based on the current knowledge and understanding of the participating beneficiaries.

The Retrieval and Segregation Toolbox Compartment may not be appropriate for seriously damaged reactors; it is assumed that reactors will have shut down as a consequence of wear and tear and economics and not due to catastrophic failure. Issues not considered by the participating beneficiaries have included fuel failure, fire or major structural damage to the reactor or its core. Specific projects are in place to recover reactors such as Chernobyl and Windscale Pile 1.

The report does not take in to account the fact that structural integrity of a nuclear reactor may vary from site to site, from design to design and as a consequence of operational history. Structural integrity will be a constraint upon process and technique selection and should be taken in to account on a project specific basis.

The report has not considered in detail i-graphite dust explosion. Research by others, especially EPRI [10], suggests that the risk of explosion is likely to be minimal. The need to consider dust explosion is considered to be a project specific requirement.

The report does not make specific tooling recommendations for retrieval and segregation of i-graphite. A number of bounding conditions and engineering prerequisites will influence the final choice.

4.2.4 Bounding Conditions / Engineering Prerequisites

Before using the CARBOWASTE retrieval and segregation tool box compartment the following information shall be established: -

1) **Delay:** The agreed reactor shutdown date and the expected delay time before i-graphite can be removed. This report has considered <25, <50 and <75 year delays.

2) **Preliminary Waste Route**: at least one route for the i-graphite and any other incidental wastes. For the implementation of retrieval and segregation only the immediate waste route at the point of dispatch from the reactor containment should require consideration. But in arriving at that implementation process, consideration may have been given during front end engineering to other specific requirements of

   a. graphite treatment,
   b. recycling and
   c. reuse or final disposal.

   Scenarios can be envisaged where these activities could take place both in-situ or after the i-graphite has been dispatched from the reactor containment. These issues are more appropriately addressed by the CARBOWASTE Integrated Waste Management Toolbox, Work Package 1.

3) **Reactor/Site Dismantling Strategy**: Retrieval and segregation of the i-graphite must be compatible with the overall strategy. Strategies are likely to be influenced by policy/regulatory, waste management, reactor-design specific, site specific and possibly stakeholder requirements. The reactor dismantling activity and associated waste disposal routes will need to consider all materials in the reactor containment and the reactor structure itself and therefore processes, working environment, access, tooling and the preliminary waste route may be influenced by factors not related to the i-graphite itself. These broader considerations may restrict the choice available for i-graphite retrieval and segregation.

4) **System Interface**: The local geometry of the plant and associated operations and processes (see Table 2) that will be undertaken before, during or after the i-graphite retrieval and segregation activity. Some of these processes will be specific to i-graphite retrieval and segregation and some associated with other activities. Processes in place are likely to include: -

Table 2 Associated Processes and Operations

a.	The control and movement of personnel
b.	The control and movement of materials and tools
c.	The control and movement of waste

d.	Environmental control systems
e.	Control, Instrumentation and Monitoring systems
f.	Emergency support systems
g.	Transport, and
h.	the interaction with other facilities and infrastructure relevant to the retrieval and segregation activity.

5) **Structural integrity:** the structural integrity of the reactor or any associated facilities or buildings that are required to accommodate and implement the i-graphite retrieval and segregation steps. Also any modifications required to these facilities. The retrieval and segregation steps are summarised in Table 3.

Table 3 i-Graphite Handling Operations

a.	Location/identification of i-graphite to be moved
b.	Inspection of i-graphite materials prior to removal
c.	Preparation of tooling to retrieve the graphite (or segregate, if done in-situ)
d.	The handling of i-graphite from source up to its point of dispatch.
e.	Any in-situ or very close to source processes applied to the i-graphite
f.	Any interim packaging (e.g. lifting baskets, bins)
g.	The preliminary waste route used to dispatch i-graphite from the reactor containment.

Interim packaging is used to facilitate i-graphite handling or support segregation operations. Lifting operations need to consider the combined weight of the maximum number of graphite blocks lifted, the basket and the lifting device. It should also be noted that other items removed from the reactor containment may dictate the structural requirements of the preliminary waste route and its interaction with the reactor structure.

6) **Graphite condition**. Data or information on this may not be fully available but information identified in Table 4 is expected to assist process selection and operation.

Table 4 Graphite and Materials Property Information

a.	Knowledge of graphite as installed condition
b.	Analysis of operational history of the reactor
c.	Records of through life sampling of i-graphite
d.	Use of predictive models available or developed on CARBOWASTE (Task 2.3 – see later report)
e.	Use of other data produced on CARBOWASTE by Work Package 3
f.	Sampling prior to and throughout the retrieval and segregation processes
g.	Consideration of the likely forms in which i-graphite might be encountered based on the above information

h.	Consideration of other materials likely to mixed or associated with the i-graphite.*

7) **Radiological conditions:** the radiological conditions likely to be encountered in the region where the i-graphite will be handled and dispatched at the time of installation and operation of the i-graphite retrieval and segregation processes is highly influential on methodology selection. CARBOWASTE has considered three scenarios, i.e. i-graphite will be retrieved and managed in <25, <50 or 75 years. Specifically for the influence on retrieval this means that the local dose rate in the reactor region will be influenced by the decay of radionuclides that may not necessarily be associated with i-graphite itself but could include for example doses from activated components in reactor containments and internals. Clearly this is reactor specific with some reactors having low doses even at <25 years. Any analysis in this report has assumed that <25 will be high dose and require processes that assume remote operation is required. 75 years on the other hand will lead to significant dose reduction and handling methodologies will have considerably more choice probably ranging from manual intervention through to fully engineered production optimised systems.

The various parameters that may affect radiological conditions are summarised in Table 5

Table 5 Factors that Influence Environmental Dose Rate in the Reactor Containment

a.	Conditions relating to specific generic reactor designs
b.	Within generic reactor designs variations due to different materials of construction
c.	Variations of these conditions due to reactor operational history
d.	Variations of these conditions during handling or storage due to the presence of other active materials
e.	Variations due to the level of contamination/impurities in the i-graphite
f.	Variations due to position within the reactor
g.	Variations even within i-graphite blocks due to geometry of block and closeness to fuel

8) **Segregation requirements**. It is necessary to know if segregation of the graphite and non-graphite material or separation of i-graphite into separate waste streams is required due to the downstream waste management requirements. If this is not required, segregation may be considered if it benefits retrieval of the i-graphite in any way.

9) **If volume reduction** of the i-graphite is required due to the downstream waste management requirements.

10) **Radiological Protection Policy**. Organisation policies may be in place that dictate the criteria used to chose between remote, semi-remote or manual application. The choice should be taken on the basis of risk assessment that is acceptable to local regulators. This report does not judge between the selection of one type of technique or another but identifies the factors that need to be taken in to account in making the choice.

4.3 Description of Retrieval and Segregation Generic Process

4.3.1 Process Analysis

To facilitate the analysis of the retrieval and segregation process a generic process has been defined that is not specific to reactor type. This flow diagram needs to be applicable to a range of processes, which based on real experience, were considered to be primarily represented by GLEEP, WAGR and Fort St Vrain (similar approach to be adopted at Bugey). The characteristics of these i-graphite retrieval processes are as follows: -

- GLEEP represents a low background radioactivity retrieval process that is carried out in air largely with manual intervention. It should be noted, however, that the retrieval processes was largely mechanised because of the large number and weight of graphite blocks to be handled. Any concern for radioactivity was related to the decision on disposal route for the graphite blocks and this impacted on the steps in the process, which in this case required the i-graphite to be shreded prior to dispatch. The i-graphite was retrieved in an ambient air environment. After defueling, i-graphite removal represented the key decommissioning activity prior to dismantling the containment (bioshield).

- Windscale AGR represents a high background radioactivity retrieval process that had to be carried out entirely using remote systems. Some earlier intervention works to gain entry to the reactor containment involved manual intervention. The process was carried out in an air environment and a predetermined preliminary waste route applied so that no deliberate segregation of waste was carried out. It should be noted, however, that the preliminary waste route allowed sorting of wastes to enable maximisation of packaging efficiency. In the case of Windscale AGR, i-graphite retrieval was only one of a number of dismantling phases to dismantle a complete reactor. Two of nine phases included i-graphite removal, firstly the removal of the thermal shield, and later the removal of the core-graphite (also some reflector graphite).

- Bugey represents a high background radioactivity retrieval process that has decided to use water as a shield during the dismantling of the reactor including removal of i-graphite. This approach will be taken because of the need to dismantle in less than 25 years post closure and to be able to use i-graphite handling techniques closer to a “hands-on” condition because of the presence of the water as a shield. The presence of the water adds a variation in that some Chlorine 36 is expected to be extracted and Bugey requires a water treatment system as opposed to GLEEP and WAGR which were concerned with airborne discharges.

As it transpires, all of these options have been undertaken in the 25 year context though for different drivers. The site was required for other use at GLEEP and maintenance cost reduction was of interest. Windscale AGR was a demonstration project for Gas Cooled Graphite Moderated Reactors, otherwise its decommissioning might have been deferred. Bugey is being decommissioned early because of a requirement by law of France to do so.

The Retrieval and Segregation Workshop sought to define a generic process that could fit these three alternative options. It should be noted, however, that a 4th process, which is largely expected to be unique but should not be ignored as an option has been applied in Germany.

- Jeulich HTGR reactor pressure vessel is filled with a light weight grout with the i-graphite (and some fuel) still in place. The whole reactor pressure vessel is then removed by heavy lift crane and transported to a new storage location on site. This represents a zero retrieval option in which the i-graphite remains in situ and it is the reactor pressure vessel that is retrieved.

Clearly, at some future time it will be necessary to consider whether the reactor pressure vessel is disposed of with the i-graphite still in place or some method is devised to separated the i-graphite from the reactor pressure vessel and/or grout. This report has not given consideration to this and the generic model assumes that i-graphite will be retrieved from the reactor pressure vessel.

Prior to consideration at the Work Package 2 workshop a basic flow diagram, Figure 1, was defined that considers the process from the point of view of a piece of graphite. (condition not stated).

Location will commence with study of reactor drawings, records both written and visual and interactive visual examination during retrieval operations.

Inspection will be specific examination of a block/component to be removed to decide if it is in a fit state to be retrieved and which is the best tool to do it.

Preparation will be the process of interfacing with and securing the block/component to be removed.

Handle is the process involving the tool/operating manipulator/lifting devices necessary to move the block or to a location where it may be operated upon prior to dispatch.

Segregate, process or interim package are optional activities that will be undertaken in any order prior to the graphite being dispatched from the vicinity of the reactor.

Dispatch is a variable and for clarification and by way of example, Windscale AGR was modified to include a waste route adjacent to the reactor pressure vessel and the dispatch was to a waste packaging and grouting station where the final waste package was produced. In the case of GLEEP a shredding plant was adjacent to the reactor containment (bioshield) and used prior to the waste being placed in and then dispatched in its interim storage container. For Bugey the reader is referred to the workshop presentation [Appendix 3].

The above processes form the main process line. There are clearly other ancillary processes that need to go on in parallel and some of these are listed in section 4.3.7 (Table 11).

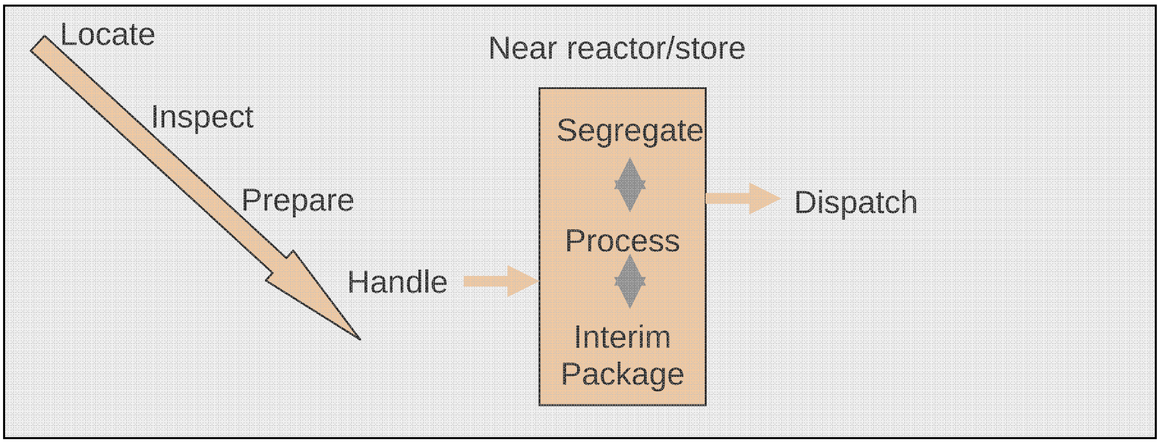

Figure 1 – Process to retrieve and dispatch a piece of graphite from a reactor or store

4.3.2 Planning Analysis

A number of stages will be applied to the resolution of the engineering outcome to this process. Stage 1 is planning, Stage 2 is the Safety Case and Stage 3 is implementation. Progressively, more knowledge and engineering detail will emerge as the project progresses through the various stages and in deed the boundaries between these stages may vary from State to State, regulator to regulator and owner to owner. During Stage 1 it is expected that the emphasis will be largely on overall reactor decommissioning of which i-graphite will be one part. It is at Safety Case stage that it is expected to start to home in on the detail of the i-graphite process. It is possible that the i-graphite safety case may be part of an overarching safety case for total reactor decommissioning. But for some projects, as was the case at GLEEP, the i-graphite removal was the prime safety case. At implementation stage, which is taken to include all design, procedural, mobilisation and i-graphite removal and dispatch activity, engineering knowledge will be required and / or available in considerable detail.

With respect to planning, in Stage 1 context it relates to the definition of endpoints, organisation, funding and strategic approach to the project. Planning in terms of delivery, optimisation and change management will continue to the completion of the process. It is very important to recognise that planning and risk management of safety, technical, time and commercial relationships are fundamental to project success. Thorough planning is important and obtaining up front knowledge of i-graphite condition and reactor containment condition will encourage project success. But despite this the unexpected will be encountered at implementation stage so flexibility of approach, tooling and skills is equally important and these will be supplemented by training and practice.

4.3.3 Key Engineering Tasks

The engineering challenges were initially considered in task 2.1 which reviewed the information in the WP1 review report, other information provided by beneficiaries and defined 4 key issues to understand for successful retrieval of the i-graphite. Table 6 lists a proposed decision making hierarchy. Steps 2 and 3 are possibly interchangeable depending on circumstance.

Table 6 Engineering Hierarchy for Graphite Retrieval Solution

1.	Process must comply with the constraints of the preliminary waste route and at least one such route must be available.
2.	The condition and status of the i-graphite and any associated components will impact upon the tooling selected for retrieving and segregating it
3.	Mode of access in to the reactor containment fundamentally affects the practicable opportunities
4.	Ancillary processes that will integrate with, or interfere with, or be required by, the i-graphite retrieval and segregation process.

Additional presentations were given at the workshop on the challenges facing the Magnox, RBMK and UNGG designs. These presentations are included in Appendix 3.

4.3.4 The Preliminary Waste Route

Once that the preliminary waste route has been defined it will have a significant impact and constraint upon the options available for retrieval and segregation of the i-graphite. Parameters to consider are listed in Table 7.

Table 7 Preliminary Waste Route Constraints

1.	Any constraints due to the waste disposal containers that will be used after dispatch of the graphite.	
2.	Any constraints that prescribe the need to segregate different materials retrieved to go to separate waste containers.	
3.	Any constraints on the rate at which graphite can be removed from the reactor containment.	
4.	Any decisions to use more than one waste disposal route due to different radioactivity levels in the graphite or associated materials.	
5.	Any constraints created by the need to manage secondary waste routes arising from, for example, the processes and working environments that might be used or any other materials introduced to facilitate retrieval and segregation.	
6.	Form of the waste is important when considering the next treatment stage. For example, for gasification the graphite may have to be in a crushed form.	
7.	The assessed initial state of the graphite pre-retrieval.	
8.	Any conditions imposed on the graphite immediately post-retrieval.	
9.	The requirements for primary routes in terms of waste forms also applies to secondary wastes.	
10.	Packing factor considerations:	
	a.	dictated by activity level
	b.	% hole space in brick/tile
11.	The need for size reduction and facilities to do it	
12.	Allow for varied approach e.g. a range of tools required	
	a.	Could be integrated with treatment
	b.	Keep it simple

4.3.5 The Mode of Access to the Reactor Containment

Various access routes are required into the reactor containment, which will need to accommodate all the process activities shown in figure 1 above together with any additional access supporting activities as identified in Table 8.

Table 8 Functions of Reactor Containment Access

1.	operators, where appropriate
2.	tools
3.	maintenance
4.	recovery of failed equipment

5.	environment control
6.	control, instrumentation and monitoring
7.	and any other activities that may be peculiar to specific reactor designs.
8.	maintenance of containment especially for underwater retrieval

For the main access route for retrieval and segregation, Table 9 summarised routes considered or used to date. Others may be envisaged as access is reactor specific.

Table 9 Potential Access Routes

1.	Removal of the head of the reactor pressure vessel
2.	Access through a plug in the top of the reactor pressure vessel or containment
3.	Other routes via penetrations in to the reactor, the most obvious being through access routes for fuel, instrumentation or other process penetrations
4.	Other scenarios could be envisaged such as via gas-circulator routes on Magnox and AGR or process channels as on RBMK (See Appendix 4)
5.	The above routes may require first removal of or access through the charge face of the reactor depending on the design

In many reactor dismantling cases it is possible that the access route may be dictated by other reactor dismantling considerations and the graphite may not be the critical activity in choosing the best access route. This will be a constraint on the choice of methodology.

With respect to the access via penetrations, Appendix 4, which describes possible options for Ignalina NPP RBMK Reactors includes review of the experience of restoring the graphite core at Leningrad NPP Unit 3. A steam-gas exhaust passage installed on top of the radiation shield of the reactor (see section 1.1 of Appendix 4) was used as the access route to remove graphite bricks and columns of bricks. The route also involved removing some of the reactor channels to clear a route.

4.3.6 Graphite and component status

From the information in the Work Package 1 Review Report [7], in the IAEA TECDOC 1521 [2] and other sources examined such as the museum at Bugey, and other information known to participating beneficiaries, there is considerable information on reactor cores and internals in the as built condition particularly for Magnox, RBMK and UNGG and specific features of other reactor designs. Whilst this detail often describes the geometry of the graphite blocks and how they are keyed together and constrained there is less information on other features such as process instrumentation, state of graphite at closure, or materials that may have been transported around the graphite circuit and deposited in or around the core area or in-deed outside the reactor containment in heat exchangers for example. So retrieval and segregation will need to recognise: -

- i-graphite and directly associated contamination or activation
- the quantities of i-graphite and other materials that need to be removed
- other associated materials, structures and components that may be in the way or integrated with the graphite that could be readily separated from the graphite
- other substances that may be encountered in the vicinity on the one hand the fluids of the working environment and on the other hazardous wastes such as asbestos.
- the condition of the graphite at the time of closure of the reactor and after any decommissioning delay.

A generic list of materials, components that might be encountered include for example:

Table 10 Reactor Graphite and Associated Components

1.	Graphite Bricks with holes
2.	Graphite Bricks without holes
3.	Spacers (e.g. the cylindrical in RBMK)
4.	Tiles (e.g. in Magnox)
5.	Zirconium pins (e.g. in Magnox)
6.	Graphite keys (e.g. in Magnox)
7.	Fuel element debris
8.	Miscellaneous debris (arising from reactor operation)
9.	Graphite dust
10.	Contamination in the graphite
11.	Contamination on or mixed with the graphite
12.	Water (either deliberate or accidental)
13.	Gases (Air, Nitrogen, Helium)
14.	Some reactors have additional components/fixings in the graphite. For example nails/plates were used for mounting the graphite in the German THTR.
15.	Asbestos
16.	Insulation
17.	Debris a. From initial penetration works b. From reactor operations
18.	Thermocouples
19.	Dropped items/tools
20.	Possibility of minor quantities of fuel
21.	Minor fuel components
22.	Cables, cable trays
23.	Merging of blocks (a condition considered to be prevalent in Bugey)
24.	Details of the geometry of keys/blocks

In addition in vaults, silos and other stores fuel element debris containing graphite may be found. A report covering this at Vandellos is included in Appendix 5. Such wastes are also known to exist at Tokai Mura Magnox, Berkeley Magnox (Fuel element debris mixed), Hunterston A Magnox graphite sleeves and on RBMK; graphite sleeves.

It is the function of Work Package 3, Characterisation to identify and expand the information available on the properties of i-graphite that might impact upon other activities in the Route Map and Work Package 1 produced lists of properties of interest including those of interest to Retrieval and Segregation. See Appendix 6.

4.3.7 Ancillary Processes

The ancillary processes necessary to make the retrieval and segregation process function effectively and safely are developed from the Data Pack and listed in Table 11.

Table 11 Ancillary processes necessary to support Graphite Retrieval

a. The control and movement of personnel.
b. The control and movement of materials and tools
c. The control and movement of waste
d. Environmental control systems
e. Control, Instrumentation and Monitoring systems
f. Emergency support systems
g. Transport
h. Interaction with other facilities and infrastructure relevant to the retrieval and segregation activity.
i. Management of dusts, aerosols and filtration systems
j. Manage any catalytic or chemical reactions that could take place in the reactor containment due to dismantling operations
k. Influence the performance of any of the above by appropriate physical modelling and training

4.3.8 Generic Process Flow Diagram

Based on consideration of the above points and debate by the workshop team a generic flow diagram was produced that should fit the majority of i-graphite retrieval and segregation projects. This is presented in figure 2, which was subsequently analysed by the workshop to define key activities within each process step. The sampling activity is a key and vital part of the process but sampling itself requires, plan, access to core, handling/removal, segregation and dispatch. So it was not considered separately, but it should be noted that there is considerable experience around Europe with respect to sampling i-graphite from reactors and this is not discussed in this report. The reader is referred to the review report in Work Package 1.

In the case of Handling/Removal, Segregation and Dispatch, these activities involve many similar and integrated operations and it was decided to review this group of activities as a whole. Many issues relating to these activities were introduced in the Task 2.1 Data Pack sections 3.3.1 to 3.3.7 and the various presentations in the workshop. The following analysis follows on from this background.

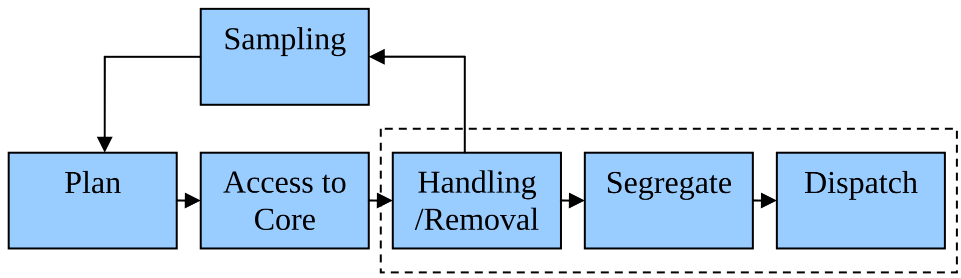

Figure 2 Generic Flow Diagram for an i-graphite Retrieval and Segregation Process

Figure 2 is considered more comprehensive than figure 1 because it considers the process steps rather than the movement of an individual piece or group of pieces of i-graphite. The top issues in each individual element are as follows.

Plan

In preparing the terms of reference for the project planning stage documentation describing the following elements shall be produced as far as is reasonably practicable.

1. Regulatory Requirements
2. Description of the Original Design
3. Relevant records of the Operational History that may impact on i-graphite condition
4. Review of Previous Experience that has been applied to access to the core, for sampling, handling and removal of i-graphite or other relevant materials, segregation of the i-graphite from these other relevant materials and all the associated management, control and environmental systems. A checklist of systems is given in Table 11.
5. Timescale/Endpoints. A clear strategy, staged as necessary should be defined that takes note of national policies or law, other strategic considerations such as resources, integration with parallel projects, business plans, funding
6. Waste Route. Available waste routes and information relating to their conditions for acceptance should be defined.
7. Optional approaches to the flow diagram in figure 1 and the constraints in 1-6 above should be evaluated and assessed to defined a “tolerance box” of processes that are considered feasible for i-graphite retrieval and segregation
8. Uncertainties (Risk Management) relating to the feasibility of the process should be identified that may lead to further development or training requirements

9. Costs should be established for the various options and in more detail for preferred processes so that financing demands can be established.
10. Use of existing infrastructure/staff/knowledge should be considered as far as possible to avoid un-necessary expenditure, minimise risks and encourage process rationalisation.

Note: There is a need to consider the total decommissioning project planning cycle. To ensure that this is done comprehensively is not in the scope of this report although reference is made at appropriate points. Further information may be obtained from sources such as the IAEA, Euratom, National Regulators, Company Quality Systems for the recommended documentation requirements at the planning stage. It is not expected that these will be specific to i-graphite.

Access to Core

Whilst it is clear that access to the core is necessary for i-graphite retrieval, there are several processes that need to be considered in designing and operating this access. These will include:

1. Access for retrieval equipment and possibly other processes associated with segregation, volume reduction or processing. These latter processes could be within the primary containment for the graphite or might be integrated into the waste export route but integrated in terms of material flow*
2. The routing of the waste needs to consider the route out, i.e. how the graphite approaches the access point, the export through the access, which may involve airlocks/gamma gates for example, and changes that might need to be made to achieve this, for example modifications to core, containment, graphite, tooling for operational reasons*
3. Reach all parts from the point of view of size of core, layout of core and obstructions, range of jigs/tools and how these are safely supported, viewed, controlled and recovered*
4. Environmental control
    a. Dose / shielding / containment
    b. Air / water / gas management and quality
    c. Dust management
    d. Sealing
5. Existing / new routes. Existing routes are those access routes to the core used during the operational phase of the reactor, e.g. process access routes (for example Appendix 4) or cooling gas ducts. New routes may be modifications to these, opportunities that arise due to management of earlier decommissioning operations in the primary containment, reverse construction routes, e.g. the removal of the reactor pressure vessel head, the use of access plugs or the creation of such plugs. The interaction of the charge face and the reactor primary containment needs to be considered, where appropriate.

**Note: Interaction with other dismantling activities and services (see section 4.3.7) should not be overlooked.*

Handling / Removal, Segregate, Dispatch

These processes have been linked together because, certainly if viewed from an optimised process engineering perspective, are likely to form an integrated process. It is conceivable, but generally thought unlikely, that buffer storage areas might be considered in a very large reactor but in general this is expected to be the loading of several i-graphite blocks or pieces in to a single handling container, for example a basket or the parking of pieces to one side that are not convenient or safe to lift with the current tool that is in use. Most of the issues below affect this entire process from handling right through to dispatch. It is possible that some integrated processes might be placed just outside the primary containment, say on the charge face of a reactor or as was the case at GLEEP, where radiation levels were low, on the floor, next to the reactor bioshield in the building that housed it.

1. Sampling / validation for risk management, health and safety and environmental management purposes; the following need to be understood or estimated: -

    a. Radiation
        i. Graphite
        ii. Activated components
        iii. Contamination
    b. Physical / Mechanical properties

2. Working environment options not only affect the working environment but depending on the post retrieval activity and waste form specification require separation of the i-graphite and the working environment material. Considering the example of Bugey, separation is not required because water will be added in any case to the encapsulation medium for the retrieved i-graphite. It is possible that some process may be developed in the future that does require separation but it is not appropriate to speculate on this. As for access, the media which might be encountered are: -

        i. Air
        ii. Gas
        iii. Water

In selecting the appropriate processes, tools and procedures the following need to be understood:

3. Physical dimensions / geometry of containment and components (including graphite and associated non-graphite components)
4. Condition of i-graphite and other components for lifting

5. Choice of Solutions (e.g. range of tools appropriate to the environment)
6. Protection → Proximity optimisation**
7. Risk / Hazard Analysis (a check list appears in Appendix 9)
8. The requirement or not to provide traceability of components removed to meet waste conditions of acceptance or to obtain new i-graphite data.
9. Mechanical / Chemical Engineering Design: -

    a. Release of components (keys, restraints, anti-lift devices, other obstructions)
    b. Graphite supporting system
        i. structure to handle tooling / loads
        ii. facilitate efficient handling and movement of i-graphite
        iii. avoid dropping or further damaging graphite (or to accept this and deal later with dropped items)
    c. Segregation operations, i.e. undertake any in-situ operations on the i-graphite e.g. physical or chemical segregation, sorting,
    d. Different techniques for keys, solid blocks, components, etc

10. Process and Logistical Engineering; especially where there a very large numbers of similar i-graphite blocks the throughput rate needs to be optimised by considering the following operations and constraints: -

    a. Size reduce in situ
    b. Interface with preliminary waste route
    c. Secondary Wastes
    d. Number of operations
    e. Transfer to export point
    f. Headroom
    g. Waste route
    h. Technology status (including reliability, integrity, durability, suitability, functionality)

**Note: Based on the Bugey plans it was recognised that there is a benefit /detriment analysis necessary to balance the need to protect workers from radiation and other hazards on the one hand and the desirability to encourage close access and good line of sight to the i-graphite on the other when undertaking the retrieval and segregation task on the other.

11. Standards and Regulatory Compliance: -

    a. Health, Safety and Quality standards associated with the engineering process***
    b. Environmental standards associated with potential releases or discharges from operation of the process.****

12. Consider any requirements that might arise as a result of developments related to WP3 and WP4.

***Note: UK regulatory practice introduces concepts such as tolerable risk and best practicable means (BPM) that need to be considered in arriving at As Low As Reasonably Practicable solutions (ALARP). Each State will apply an appropriate regulatory standards which might use different terminology. Some information on regulatory systems in various countries is included in the Work Package 1 Review Report.

****Note: In the case of air environment the issue of dust and aerosol management was noted in the workshop as a key issue and in the case of HTGR in Germany an unusual stone / graphite material has been encountered that is subject to significant activation. By contrast the French approach to use water as the working environment will eliminate the dust problem though creating a different requirement for filters and ion-exchange resins necessary to maintain the clarity of the working environment and remove any isotopes entering the water through leaching. This has been discussed in a paper to the French Nuclear Society [11].

5.0 Use of the toolbox retrieval and segregation compartment

5.1 CARBOWASTE Retrieval and Segregation Solution Route

The concept of CARBOWASTE is to review i-graphite management as an integrated process from identification of the i-graphite status, through retrieval and treatment to its final disposal or re-use. It is therefore essential to review this integrated approach before engineering a retrieval and segregation solution. This is no different to any decommissioning good practice where strategies and waste management opportunities are reviewed before proceeding to detail design specific tasks. The hierarchical approach to resolving retrieval and segregation is shown in figure 3.

The first step of the user will be to consider retrieval and segregation in the context of the integrated waste management approach using the approach of Task 1.6 or Work Package 1. (see the blue box in Figure 3), which is related to the Multi-Criteria-Decision-Analysis (MCDA) being developed by Work Package 1. The output of the first step will include the specification of the opportunities and constraints that will need to be addressed in detailing the retrieval and segregation engineering solution.

The second step of the user will be to use the retrieval and segregation tool box compartment exemplified to develop more detailed retrieval and segregation processes and specifications. (see the green box in Figure 3). The segregation and tool box compartment is exemplified by this Work Package 2 report supported by the other Work Package 2 report on diagnostics and modelling, when this is available (the rose box in Figure 3). The preferred retrieval and segregation engineering solution needs to be iteratively reviewed to check its compatibility with the integrated waste management process (see the yellow box in Figure 3).

The third step of the user, which is beyond the scope of Work Package 2, will be to detail develop a solution (see the white box of Figure 3) and then review (see the orange box in Figure 3) the proposed solution against the integrated waste management requirements specific to the national, regulator, operator, environmental and stakeholder context. This review is represented by the thick black dotted line in Figure 3.

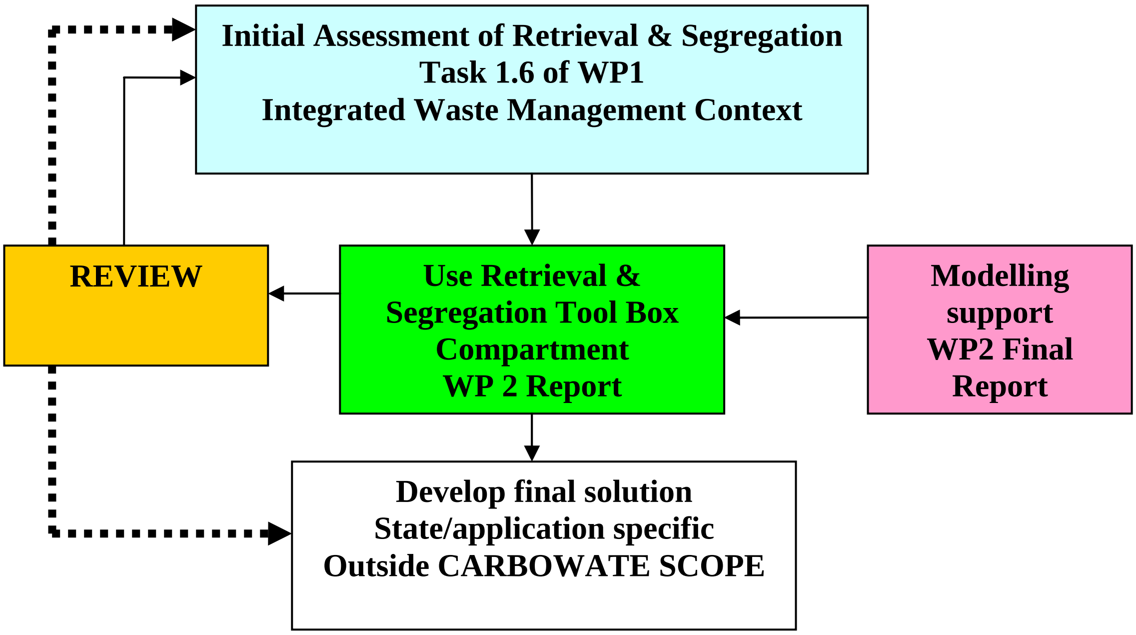

Figure 3 – Hierarchy of the Retrieval & Segregation Toolboxes

This report has been prepared ahead of the completion of the CARBOWASTE project Work Package 1 and it is important to be aware that further developments are likely to arise as the CARBOWASTE project progresses. Firstly, these will impact upon the MCDA activity in Work Package 1 that will be reported at the end of the project. Secondly, Work Package 2 is continuing until year 3 of the project with developments in the area of diagnostics and modelling. When that work is completed, this report will be reviewed and updated to ensure that links to the diagnostics and modelling are appropriately made.

5.2 The CARBOWASTE Route Map

5.2.1 The Integrated Waste Management Process

The route map (Summarised in Figure 4) has been developed in Work Package 1. For outline purposes only, a brief review of the route map is given in the following paragraphs. For greater detail the reader is referred to reports issued by Work Package 1[7].

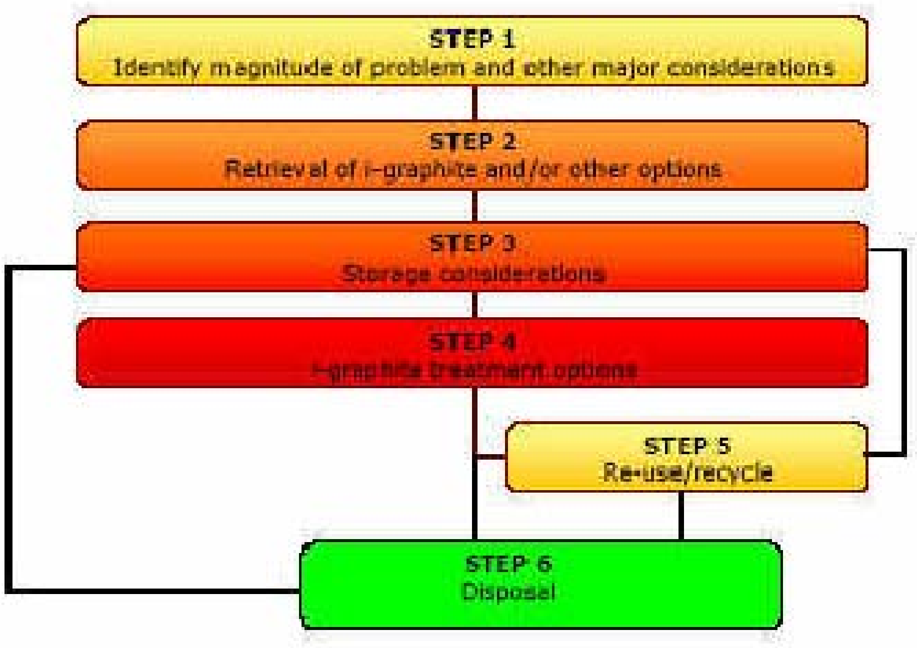

Figure 4 – Irradiated Graphite Route Map – level 1

Figure 4 shows the various steps and is extracted from Work Package 1 document [9], where more detail of each step may be found.

5.2.2 Step 2 Retrieval of i-graphite and /or other options

Within the route map step 2 is described as shown in Figure 5.

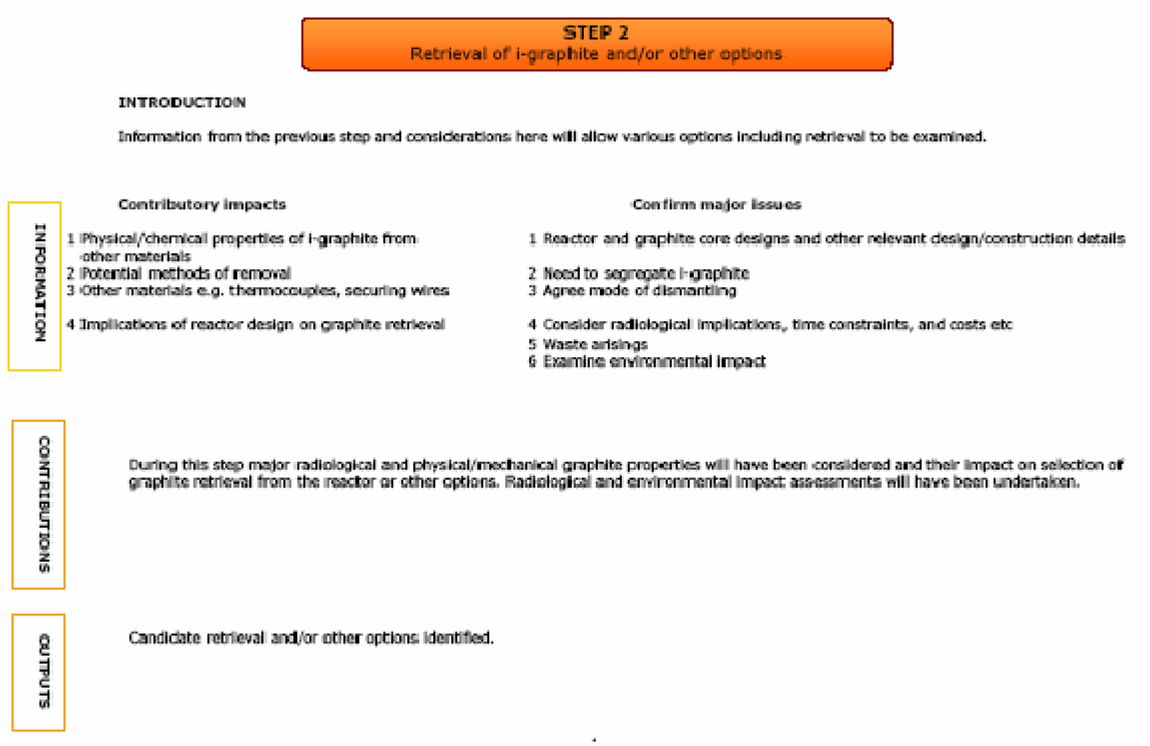

Figure 5 - Retrieval of i-graphite and/or other options

The requirements for retrieval and segregation as viewed from the Integrated Waste Route are not the same as the requirements as seen from the point of view of the retrieval and segregation engineer and some interaction is required between those responsible for developing waste routes and those charged with retrieving the i-graphite.

To better understand this task 1.6 of Work Package 1 was undertaken jointly by Work Package 1 and Work Package 2 beneficiaries. It was discovered that there are potential clashes between the definition of retrieval and segregation criteria as observed from the point of view of the integrated waste solution and as seen from the point of view of the retrieval and segregation engineer. It is recommended that a top down solution approach is therefore adopted for many retrieval and segregation requirements as outlined in figure 3.

5.2.3 Upper Level Criteria for Retrieval & Segregation

Work Package task 1.6 defined the upper level criteria for retrieval and segregation as presented in Table 12. The table is generally self explanatory, however, a number of comments need to be made that will impact upon the design of the retrieval and segregation process.

With respect to the Disposal criteria, a decision must be made with respect to the need to treat (in essence any treatment is considered to be a segregation process) i-graphite as an integrated part of the retrieval process either in the reactor containment or its extended containment, prior to dispatch (see figures 1 and 2). If the disposal strategy/mode dictates that no segregation is required, then the graphite will be removed from the reactor containment as found, and placed directly, possibly with some size reduction to facilitate waste packaging efficiency, with or without furniture into the final disposal box for which only one available waste route is assumed. This can be considered a base case and historically is represented by the example of Windscale AGR.

If the disposal strategy/mode requires that segregation is required then retrieval and segregation tools that are required to effect this may need to be challenged with respect to their feasibility, flexibility and value of undertaking so that a decision can be made as to whether this segregation activity is best done prior to or after dispatch. GLEEP represents an historical case where i-graphite was shredded very close to the reactor containment to facilitate downstream operations. It should be noted, however, that not all i-graphite was shredded in-line, but some was temporarily contained and removed to a buffer store to avoid segregation causing critical path problems.

Being upper level criteria, a relatively simplistic view is taken, and in advising the waste management process on what is feasible, flexible and of value, the retrieval and segregation engineer is likely to have to consider: -

- Selective removal of i-graphite from different parts of the core taking account of variations of activity levels within the core. It is generally understood that maximum radioactivity levels are in those blocks nearest to the centre of the core.
- Sort graphite outside in an extension to or outside of the reactor containment prior to dispatch.
- Segregation by splitting of graphite blocks into parts of different specific radioactivity. It has been reported at Manchester University Graphite Training Course that the radioactivity level within individual moderator blocks decreases with distance from the fuel channels
- Volume reduce the graphite prior to dispatch to suit an alternative waste route (as at GLEEP) or to fit into waste disposal containers

Table 12 Upper Level Criteria for Retrieval and Segregation

STAKEHOLDER

RISK

ENVIRONMENTAL

RETRIEVAL
- Reactor/core design
- Delayed storage
- Quantity of graphite

TREATMENT
- Flexibility
- Efficiency

DISPOSAL
- dedicated repository
- on site disposal
- repository boundary limits (size and radioactivity)

Retrieval of i-graphite
- Delay store of 25, 50 or 75 years;
- Reactor/core design;
- Quantity of graphite to be retrieved and rate;
- Segregation required of graphite based on ^{14}C content and/or other non graphite materials.

Removal of ^{14}C
- Process requires particulate i-graphite;
- Process capable of handling massive i-graphite;
- Above processes incapable of accommodating non-graphite material;
- Process capable of handling wet/dry i-graphite;
- Minimal shielding of ^{14}C removal process.

Consideration of disposal strategy/mode
- No treatment to remove ^{14}C and hence ILW repository;
- Treatment to achieve LLW;
- Relaxed ^{14}C authorisation that requires no ^{14}C treatment;
- Volume reduction required for above three options;
- Minimal shielding of i-graphite conditioning (cement encapsulation) process [assumes encapsulation needed for all options].

Top level Retrieval issues

Timing	State of Graphite
Quantity of Graphite	Reactor design
Local Infrastructure	Politics
Social acceptability	Preliminary waste route
Skills and resource	Quantity of others (non-graphite)
Availability	HSSE

- Note that in most reactor containments the core does not comprise solely graphite but also many other components and substances either necessary for construction stability, operational control of the reactor, operational contamination, cross-contamination caused by other in reactor dismantling activities or as a working environment in which reactor dismantling takes place.

- It is implied that the detail of information for waste characterisation and sentencing at the time of retrieval and segregation will probably require greater depth if multiple waste routes are used than if a single waste route is used.

With respect to the treatment criteria, they may be of interest for the application of treatment before and after dispatch. The CARBOWASTE project and other earlier research or concepts such as incineration or gasification of graphite have generally considered treatment after dispatch. GLEEP, see above, is an anomaly as it was both outside the reactor but prior to dispatch but in a low activity environment so that choice was not inhibited. However, at recent EPRI workshops that have been held in association with the CARBOWASTE project, at least two proposals have been described that link the retrieval and treatment activities. In challenging the retrieval and segregation tools with respect to their feasibility, flexibility and value the following segregation options may need to be considered: -

- In-situ size reduction prior to removal from the reactor containment
- In-situ decontamination of graphite dry
- In-situ decontamination of graphite wet (incidental). This may be an outcome of the proposals to decommissioning the French UNGG reactor at Bugey underwater.
- In-situ decontamination of graphite wet (deliberate)
- Size reduction of graphite removed from the reactor containment but prior to dispatch. This could be in an extended reactor containment or very close by as at GLEEP.
- Environmental control of any C-14 etc that is discharged from the graphite due to in-situ or pre-dispatch processes.
- The segregation of non-graphite materials e.g. instrumentation, keys and pins, secondary materials introduced to control the environment or to operate the segregation process.

Retrieval is the prime activity as far as the engineering of the retrieval and segregation process is concerned. Whilst the requirements noted above for disposal and treatment may need to be engineered into the retrieval and segregation process, no graphite will be able to leave the reactor containment unless the issues in retrieval are fully addressed as follows.

The delay period will, in most cases, significantly impact upon the methodology and operating process used. There will be a tendency towards fully remote or fully shielded environments (e.g. underwater) as the delay is minimised and an opportunity to move towards semi-remote or even some manual operations as the delay period extends.

The reactor containment and core design are fundamental considerations as this will impact upon the mode of access into and out of the reactor containment, the state and conditions inside the reactor containment once access has been achieved and the components likely to be encountered. It might be expected that common generic assumptions can be adopted for the different reactor designs such as Magnox, UNGG, RBMK and in the future AGR. But in practice significant variations arise with in common designs because it must be remembered that legacy plant was constructed in the relatively early days of the nuclear industry and continual design variation and improvement exists both from the perspective of different manufacturers of plant and due to development improvements. For example reactors have both steel and concrete pressure vessels, different orientation of the primary circuit cooling systems, different shapes pinning and constraint of the core. To this should be added the caution of different operational histories.

The quantity and rate of graphite to be retrieved is an issue that will be perceived differently by waste disposal route operators and retrieval operators. In the case of the disposal route, the quantity will impact on several downstream processes such as interim storage, transport systems, availability of processing capacity and waste disposal site capacity. There could be possible hold-up if the downstream facilities or transport routes become unavailable during processing or impact on plans if these facilities or routes change over long periods of time. The retrieval operator, on the other hand, once the retrieval process has been given the go-ahead will be keen to avoid bottlenecks so that operational equipment and labour resources are well utilised, that equipment of adequate performance and reliability has been chosen compatible with the number of operations to be undertaken in the working environment.

The last retrieval issue, segregation is important from the perspective of conditions of acceptance of i-graphite by the various downstream treatment or disposal processes. Separately, from the perspective of the retrieval engineer, segregation of some materials may be a necessary activity to facility the logistics of the retrieval process

5.2.4 Developing Criteria for the Work Package 2 toolbox compartment

The boundary between Step 1 and Step 2 needs to be considered carefully and has been investigated by a group from the Work Package 1 and Work Package 2 teams (NNL, Doosan Babcock and AMEC). This resulted in a review of the information in the Table 12 at a workshop on 11th November 2008 and this led to the identification of input data for the Work Package 2 workshop, the information being reported in Task 2.1 Data Pack [8]. Firstly, the principle functions of the Retrieval and Segregation activity were defined and secondly, groups and factors influencing retrieval and segregation criteria were identified by standard workshop brainstorming techniques.

The key processes that need to be feasible, flexible and add value for the retrieval and segregation activity are listed in Table 13.

Table 13 The key sub process of the Retrieval and Segregation process

1	The choice of process (from figure 2) and managing it in the chosen environment (air, water or any other suitable environment
2	The access to the reactor containment
3	Handling the graphite in accordance with its detailed condition, position and interaction with associated plant and materials.
4	The preliminary waste route.

The groups and factors influencing the above activities are listed in Appendix 7 and were grouped under the following headings.

1 Political/Social/Economic
2 Timing
3 EHS&Q
4 Waste Management
5 Reactor Containment / Core Design

Appendix 7 may be considered a checklist that can be used or expanded by users as appropriate to their specific needs. Following, this initial analysis NNL developed further criteria for use in the Work Package 2 workshop and this will be discussed in section 6.

A further criteria may be added, which is of general nuclear interest but never the less of concern in the current terrorist climate: -

6 Security

5.2.5 The Preliminary Waste Route

The key link between the Integrated Waste Route and the Retrieval and Segregation Process has been termed the preliminary waste route.

Figures 1 and 2 show the output of the retrieval and segregation process as “dispatch”. This is defined for retrieval and segregation as the point at which the i-graphite and any other associated materials are considered to be removed from the reactor containment and are in a state that will enable subsequent packaging (possibly including grouting) or further conditioning or treatment. Even this definition is open to interpretation because the envelope of working area of a reactor containment may be different in the decommissioning phase compared with what it was during the operational phase.

An alternative definition, taking a production engineering view, is that the point of dispatch to the downstream waste route is at the end of the integrated waste retrieval, segregation and associated handling activity such that any bottleneck in any part of that process would restrict the rate at which i-graphite could be retrieved. Thus the retrieval “production line” prior to up to the point of dispatch could comprise those listed in Table 14.

Table 14 Retrieval and Segregation Production Line Components

1. Activities within the region of the reactor containment as it was during its operational lifetime
2. An extension to the reactor containment that is added specifically to manage the i-graphite immediately after its retrieval
3. Posting systems that allow exit of the i-graphite from the containment defined in a) and/or b)
4. Any other operations that may be in-line with 1) and/or 2) and/or 3) and located very close to the reactor containment for example on the pile cap or adjacent to the reactor containment and provided these operations are before any downstream processing operation that is not specifically necessary to effect retrieval of the i-graphite.
5. Buffer storage or flexibility built in to the activities 1) to 4) above.

It is stating the obvious that to enable retrieval of i-graphite at least one preliminary waste route must exist. Also because this preliminary waste route may be used for other decommissioning activities, either before or after the i-graphite has been removed, the preliminary waste route cannot be designed considering i-graphite in isolation. It has to be specified as part of the overall facility decommissioning plan and within this the requirements of i-graphite need to be considered. This is probably best done in Step 1 (see blue box in figure 3). The contributory impacts and major issues, listed in figure 5, will need to be addressed.

When considering the preliminary waste route, it is should be noted that the requirements of it could vary with time. In this context it is noted that environmental radiation levels, infrastructure condition/availability and previous decommissioning activity within the reactor or the site on which it is located are functions of time.

Once the preliminary waste route has been defined it may have a significant impact upon the options available for retrieval and segregation of the i-graphite. Some Constraints that may have significant impact are listed in Table 15.

Table 15 Preliminary Waste Route Constraints on Retrieval Process

1. The design of waste disposal containers that will be used after dispatch of the graphite.
2. The need to segregate different materials retrieved to go to separate waste containers or waste routes.

3. The rate at which graphite can be removed from the reactor containment – see also Table 14
4. The selection of waste route based on radioactivity levels in the graphite or associated materials, or due to other properties of the waste that may not necessarily be due to radioactive properties, e.g. asbestos
5. Management of the secondary waste routes arising from, for example, the processes and working environments that might be used, or any other materials introduced to facilitate retrieval and segregation.

In conclusion, the preliminary waste route represents a "gate" (see Appendix 1) that limits the progression from the upfront retrieval and segregation activity to downstream waste route that may include Steps 4, 5 or 6 of the CARBOWASTE Route Map. It is therefore extremely important that the preliminary waste route is resolved at an early date and adequate time and investment is devoted to getting this right first time.

Whilst this report is not concerned with the down stream processes, those developing and designing down stream processes such as treatment, recycling or disposal will do well to be aware of the practicalities and limitations of the retrieval process in so far as it places any constraints upon the input streams to their processes. For example if the retrieval and segregation process finds it difficult to guarantee an output stream specification that is required for a downstream process, the downstream process may have to build in additional segregation activity to rectify this or modify its operational parameters to deal with off-specification material.

It is clearly a two way process and the importance of CARBOWASTE Step 1, integrated waste management process cannot be understated.

5.3 Develop Selection of Preferred Retrieval and Segregation Process

5.3.1 Define the problem

The primary constraints for retrieval and segregation will have been defined as a result of the activity in Step 1 (see blue box in figure 3 and the process discussed in sections 4.1 and 4.2 above). The preliminary waste route(s) will have been defined at least in the case of un-revocable constraints upon the retrieval and segregation process on the one hand and flexibility (options) on the other.

The retrieval and segregation activity then needs to be considered in three parts: -

1) The retrieval and segregation process defined in Figure 2:

i. the technical function and quality of output of the process

ii. the logistics and production rate required
iii. the associated processes with which retrieval and segregation has to interface

2) The engineering specifications of the engineering tasks or unit operations within the process

3) The selection of the tooling to:

i. interface with the i-graphite (including and segregation activity)
ii. move tooling and the graphite to and from the point of posting and/or dispatch and handle or operate any supporting EC&I, services, environmental and safety control
iii. post the i-graphite at the dispatch point.

5.3.2 Planning Approach

The first point to consider will be the timing of the retrieval and segregation operation after shut down of the reactor. This will in most cases dictate the radiation conditions in which the retrieval and segregation activity will operate and dictate the requirements for remote operation or shielding. An example of the problem definition approach as applied to the Bugey Reactor was presented by Francis Berenger of EdF at the work package 2 workshop and his presentation is given in Appendix 3.2.

The basic principle of the EdF approach is to try and optimise level of risk versus the investment to do it. Having decided on timing, the key constraint on retrieval is the detail of the reactor design, its post closure characteristics and the condition of the graphite within it. The decommissioning planning requires not only safety assessment and review of the technical challenges but also review of decommissioning efficiency. Table 16 highlights the key issues that need to be resolved to overcome operational risks.

Table 16 Upper Level Approach to Reducing Operational Risk

1. How to trap contamination
2. How to facilitate radiation protection
3. How to ensure good visibility of the work zone.

The first two points, 1) and 2) are primarily concerned with safety but contamination spread is also a serious reason for interfering with operations and need for maintenance; 3) is primarily about operational efficiency as good visibility leads to reduced delay and uncertainty. The opportunity for an underwater environment was recognised as addressing all three of these issues but a choice scenario matrix was used and is summarised in slide 4 of Appendix 3.2. This compares the key issues for either using water as a shield (but one through which visibility can be achieved if conditions are controlled) or undertaking the activity in air using remote operations. The density of components within the reactor pressure vessel proved to be criterion and the importance of interfacing with the overall decommissioning process was also important because of the possibility of adapting tooling used on general reactor decommissioning for retrieval of i-graphite.

EdF have in the past placed considerable importance on understanding the condition of the i-graphite in its reactors and such knowledge is very beneficial to the designer of the retrieval and segregation process. Very little sampling has been done in the UK for example and designers will start with considerably less information. It is anticipated that the results of Work Package 3 will significantly add to the knowledge base.

The workshop noted for important differences between the Windscale AGR graphite condition and that in the UNGG. In the former case it was noted that shrinkage of i-graphite blocks was of assistance to the block handling whereas at Bugey it is anticipated that blocks may have merged and studies have been undertaken by EdF to release restraints and identify “key” blocks that once released would free up the majority of the neighbours.

Next, bearing in mind what has to be delivered to the preliminary waste route the estimated production rates for the i-graphite retrieval process should be considered not only with respect to the retrieval process itself but in terms of the availability of the waste route and services. It has been the practice in some countries to put a legal deadline date for the decommissioning of a nuclear reactor or in others to have planned completion dates that are set according to priorities.

Table 17 lists the parameters that the Work Package 2 workshop considered to be inputs to Planning.

Table 17 Planning Parameters

1.	Regulatory Issues
2.	Original Design of the Reactor
3.	Operational History of the Reactor
4.	Previous Experience of Retrieval and Segregation of i-graphite and other core materials
5.	Timescale / Endpoint for the Retrieval and Segregation activity
6.	Waste Route (both the preliminary waste route and other downstream influences)
7.	Uncertainties (Risk Management)
8.	Cost
9.	Use of existing infrastructure / staff / knowledge

To take forward figure 2 to become the outline design for the retrieval and segregation process, the functional requirements of each process step should take account of the issues raised in the following sections and summarised in Table 18.

Table 18 Developing the Generic Retrieval Process for Specific Applications

Section 5.3.3 Waste disposal	*What is the retrieval and segregation end point?*
Section 5.3.4 Dispatching the Graphite	*What needs to be done to achieve the end point?*
Section 5.3.5 Access to the Reactor containment	*What is the route to get to access the graphite and dispatch it to the preliminary waste route?*

The development of the i-graphite will also progress through a number of stages or review gates as summarised in Table 19.

Table 19 Gated Process for Developing the Retrieval and Segregation Process

The Planning Stage	*Step 1a Integrated Waste Management MCDA* *Step 1b Specific Retrieval and Segregation planning – including Sampling, Modelling, Data acquisition on plant design, operational history and expected i-graphite and environmental properties.* *Initial engagement with implementation supply chain to assist development of implementation specifications with appropriate risk management.*
The Safety Case Stage	*Front end and conceptual design activity in sufficient detail to enable Safety Cases and possibly functional specifications for tendering activity*
The Implementation Stage	*Detailed development and / or design of the various activities to enable an implementation team to proceed.* *1) Finalise agreed processes and backup* 2) *Build development rigs and mocks for tooling selection and validation of graphite retrieval and use of common methods and technologies used on other decommissioning tasks* 3) *Plant preparations for access to reactor and preliminary waste route noting this may already be covered for other decommissioning tasks on the reactor* 4) *Mobilisation and training of workforce to practice installation of any services and retrieval and recovery procedures for graphite.* 5) *Retrieve the graphite* *6) Quality Assurance / Records*

5.3.3 Consideration of waste disposal (after Dispatch)

The timing of the retrieval and segregation needs to be linked to the availability of waste routes, which is likely to be a major constraint of the preliminary waste route. In the case of Windscale AGR, GLEEP and Fort St Vrain such preliminary waste routes existed although very different in each case. Whilst Bugey has not yet been retrieved, the proposed solution is again based on a defined planned waste route. It is useful to consider the diverse routes used by these projects. Clearly other routes may be considered including, for example, the outputs of Work Packages, 4, 5 and 6 of CARBOWASTE.

Summarising the information in the review report of Work Package 1, France is proposing a dedicated i-graphite repository, this has not yet been constructed and its availability will influence the start time for reactor dismantling; Windscale AGR had an interim store and a letter of compliance from the disposal authority (at that time NIREX but now the Radioactive Waste Directorate of the Nuclear Decommissioning Authority), which did not require any segregation as all waste and associated furniture went in to the same 2x2x2m cubic concrete waste boxes that were subsequently grouted and sent to interim store. GLEEP had the choice of a low level waste route (packaging and grouting of as retrieved graphite blocks) or a thermal processing route (that was used) that required the i-graphite to be shredded to <50mm and separated from any other materials within the reactor containment. (Other materials were not a significant feature of the pile design although some asbestos that was found required specialist handling); Fort St Vrain's graphite was removed as part of the fuel route, only reflector graphite being part of the decommissioning process.

5.3.4 Dispatching the Graphite

Considering the waste disposal approaches discussed in section 5.3.3, others that could be developed in the future, Work Package 1 has listed options for upper level disposal strategy / modes as indicated in Table 12. These downstream issues will impact upon the preliminary waste route and the constraints have been listed in Table 15. The Work Package 2 workshop considered the upstream retrieval and segregation issues and it was concluded that the operations of handling / removal, segregation and dispatch (see dashed line box in figure 2) needed to be considered in an integrated manner as they would not be separated in practice.

Table 20 lists the input parameters and requirements for Retrieval to Dispatch activity and further discussion and explanation of the parameters in the table follows in b) to k). Paragraph a) points out how consideration of these parameters needs to be linked to various engineering and gated processes.

a) Application of the Retrieval and Segregation Stages

The design will follow a phased approach; viz. process specification; then engineering specification, and finally tooling selection. Points 1-20 in Table 20 need to be addressed. The phases are not independent but are integrated so that some knowledge of tooling experience is desirable when the process is being specified and the process may need to be flexible to adapt to different tooling needs during implementation. Most of the above points are self explanatory; some additional information and review is given in the following sections. In addition the application of some of points 1-20 may be considered at each stage of the gated process referred to in section 5.3.2 and Table 19. A detailed review of technology selection is reported in section 7.3.

Table 20 Handling/Removal, Segregation and Dispatch Parameters

1. Sampling / validation of condition of components to be retrieved (see b) a. Radiation i. Graphite ii. Activated components b. Physical/Mechanical
2. Process design of the working environment (see c) a. Local management and control i. Air ii. Water iii. Other e.g. gas b. External systems i. Inventory control; feed and bleed, storage ii. Circulation iii. Condition management; purification, temperature, clarity
3. Physical dimensions / geometry (size and position of components)
4. Condition relevant to a range of lifting methods (physical condition and geometry)
5. Available solutions (the range of viable tools)
6. Balancing worker protection and proximity to the workface (see d)
7. Risk / Hazard /Analysis (see e)
8. Traceability of components especially for waste route verification
9. Mechanical Design a. Release of components (keys, restraints, anti-lift devices, instrumentation) b. Graphite supporting system; structures, manipulators, tools c. Associated EC&I systems d. Obstructed areas (see f)
10. Size reduction in situ (see g) a. Required to meet integrated waste management requirements, or b. Required to meet retrieval process requirements
11. Secondary wastes introduced
12. Number and frequency of operations (see h)
13. Transfer to and through the dispatch point
14. Headroom
15. Preliminary waste route constraints
16. Technology status; performance, reliability, maintainability, complexity (see i)
17. Relate process and tooling selection to the Best Practical Means (see j)

18. Different techniques for keys, solid blocks, components, etc
19. Waste packaging efficiency (see k)
20. Grout in situ (see l)

b) Sampling and Validation

Work Package 1 produced lists of data requirements (characteristics / properties) [12] for retrieval and segregation to be investigated in Work Package 3. Further information on the relationship of C-14 in graphite is also being discovered in work in Work Packages 3 and 4.

A sampling loop is illustrated in Figure 2. The logic behind this loop is as follows.

It is to the benefit of the retrieval and segregation process if the maximum information is available at project start. This might come from operational sampling records, modelling based on data from other projects and / or calculations based on reactor operational history.

It is known, however, that such data is scarce and hence the requirement for additional information from Work Package 3.

It is likely that additional sampling may be considered by the retrieval and segregation team at the beginning and during the project. Table 21 gives a number of reasons why this data will be useful. Points 3 and 4 are particularly important. The first is necessary to ensure costs are minimised and the project proceeds to good time with further reduced harm to the operators and others. The second is essential for quality assurance purposes. The project could be at risk if waste and waste route are not validated.

Table 21 Requirements for Sampling

1. Design the retrieval and segregation process
2. Select the tooling.
3. Help the process to be refined and streamlined as the work proceeds
4. Provide valuable data for waste management authorities

If such specific reactor intervention for sampling is required then it is considered that the same flow diagram applies to sampling as applies to retrieval and segregation. The sampling process needs to planned, access to the core area arranged, the sampling devices handled and moved, samples segregated for analysis and samples dispatched. Some lessons learned for sampling will be known to those operators who have taken samples though out the operational lifetime of the plant. The current level of published information has been recorded in the Work Package 1 review chapters and additional information has been presented by some CARBOWASTE team members at recent EPRI and IAEA conferences on graphite[3, 4, 5, 6].

c) Working environment

A logical approach to choosing the best working environment was outlined in section 5.3.2 in relation to why EdF chose water for Bugey decommissioning.

With respect to the local environmental control an important design issue will be mobility of i-graphite within the containment whilst it is being retrieved. The key issues to be resolved are the need to filter air in the dry scenario and the need to maintain line of sight visibility in the water scenario.

In the case of in-air, dust management is the challenge; using tooling that minimises dust formation, avoidance of clogging tools and instrumentation, retrieving dust collected at points of deposition and disposal of HEPA filters or other dust suppression systems. For under-water, whilst dust will be kept under control, water transparency quality must be maintained by using external water treatment plant with filtration as a minimum.

Experience of fuel cooling pond operation has suggested that water clarity might also depend on ambient weather conditions. EdF did not anticipate problems and it was noted that Fort St Vrain had been successfully decommissioned under water using similar techniques but not necessarily replicating i-graphite retrieval.

There is no experience recorded of using water sprays for controlling graphite dust. Water spray from a fireman's hose was successfully deployed at GLEEP but on concrete bioshield dismantling; it was not required for i-graphite handling, which was by drill and tap tooling, a non-aggressive process producing a small local heap of dust that could be collected by vacuum cleaner for sample analysis. There was a need to protect HP monitoring equipment from the dust.

It has been suggested that high levels of dust were collected in HEPA filters at Windscale AGR by UKAEA leading to the disposal of significant numbers of filters in the same disposal boxes as the i-graphite. No reports of this making the retrieval process particularly difficult have been reported.

An issue identified associated with German nuclear plant containing i-graphite is the presence of a form of graphite containing "stone" that can become activated and cause radiation concern. So clearly the impact of failure to control dust will be one of a number of factors that influence the choice between dismantling in-air and under-water.

Other proposals have been made at two EPRI Graphite workshops. The first suggested the use of very high pressure liquid nitrogen jets to break up the graphite and the use of foams (not necessarily in-situ) to transport the resultant i-graphite particles. Another, approach being considered is nibbling of i-graphite and vacuuming out particles in the size range 1-50mm. Processes of such type or others that might be considered may lead to a variety of environmental conditions in the reactor containment (and any extension to it) and whilst in-air and under-water retrieval are seen as the most likely approaches it is conceivable that a gas atmosphere could be considered. Although, not necessarily an option for retrieval it is noted that RBMKs operated in a Nitrogen/Helium atmosphere.

External systems are required because the working environment is likely to be circulated in and out of the reactor containment or discharged and the radioactive or any other particles carried with the medium need to be managed. Radioactive or other particles, if present, require collection and safe disposal and the quality of the medium itself must be maintained. In the case of some radioisotopes that have affinity with water, especially Cl-36, the environment may double as a partial treatment medium for the i-graphite thus affecting some level of decontamination. This will also impact upon the design specification of the external plant.

Irrespective of the choice of environment a number of questions need to be addressed: -

1) Does the environment encourage the control of dust?
2) Does the environment encourage environmental protection of the workforce?
3) How is the environment quality managed?
4) Is the environment required for an in-situ segregation process, i.e. the removal of contamination from the graphite.
5) Less likely to be considered; is the environment deliberately used to affect the mobility of the i-graphite?

d) Worker Protection

One of the main reasons for choosing underwater dismantling at Bugey is the opportunity to get the worker force close to the work point so that they easily see the i-graphite and use relatively simple manually directed tools to interact the tooling with the i-graphite.

Water is very useful because it acts both as a shield but being transparent can give line of site without the need to resort to complex CCTV systems. To maintain line of site within water requires avoidance of diffraction of light, which can be achieved by viewing vertically to the point of working. This approach contrasts with the challenges of using remote CCTV viewing, probably in more than one axis when working in-air. This type of remote viewing requires operator skill but was successfully achieved at Windscale AGR.

The line of site approach at Bugey will be feasible because of use of a working platform that will be lowered as the work proceeds and water level is reduced – the Fort St Vrain approach. The need to remove particles from the water to facilitate clarity will also benefit dose to operators by ensuring avoidance of radioactive particle build up in the surface of the water. Based on the Fort St Vrain experience, EdF do not anticipate problems of circulation of water caused by natural circulation or tool operation. Underwater operation in the nuclear industry is well established. No large temperature gradients are envisaged but never the less should not be overlooked when designing an underwater system.

e) Risk / Hazard Analysis

It will be part of the design process for retrieval and segregation to undertake HAZAN/HAZOP of the proposed designs and these will be specific to application. It is recommended that the process of Hazard and Operability Study follows any standard practice for this technique, which may be a standard internal procedure of the organisation who manages the graphite.

At an early stage consideration will be given to the materials used, the likely operations to be used and the special relationships between the operations and any associate plant built to assist the operation or located close by that might interfere with it. The potential major hazards that may arise are indicated in Table 22.

Table 22 Potential Major Hazards

Fire	Radiation	Mechanical Failure
Explosion	Noise	Heat
Detonation	Vibration	Drowning
Toxicity	Noxious material	Working at Heights
Corrosion	Asphyxia	Confined spaces

In addition to the impacts on personnel, mainly workforce but possibly the visitors and the public there are the similar impacts on the environment.

An issue that has caused great debate within the i-graphite community has been the possibility that it might combust or lead to a dust explosion. It has been of interest not only for hazard reasons but because some have proposed that i-graphite should be incinerated and this has both been demonstrated as possible under appropriate conditions and also shown to be exceptionally difficult to achieve. R&D has been undertaken by EPRI to evaluate the risk of i-graphite dust explosion and presentations have been given at several recent decommissioning workshops. At the recent workshop in Hamburg[5], Wickham concluded that it is extremely difficult to create a graphite dust explosion. Graphite is in fact an extremely inert material and the Windscale Pile fire showed this to be the case. It is also known that hot dismantling work took place in Windscale AGR without serious impact upon the graphite.

In reviewing the major hazard in Table 22 it should be born in mind that CARBOWASTE covers a wide range of time scenarios for reactor decommissioning from <25 to 75 years and some research reactors do not necessarily have very high radiation levels. It is therefore possible that for some reactors man access will be undertaken, either for operational or for set up reasons. So some risks can vary significantly with reactor design and timing. A good example of toxicity is that asbestos was encountered during the removal of the graphite from GLEEP. It could be that in low energy reactors conventional hazards outweigh radiation hazards.

The Data Pack prepared for Work Package 2 considered some of the hazard and operability topics that might arise from operations encountered during retrieval and segregation. The review cannot be considered complete as HAZOP needs to be applied ultimately to a specific design. The topics identified are included in Appendix 9. The list is provided to seed the process and as a checklist. The use of HAZOP procedures that are recognised by the industry using trained personnel must be used. However, an excellent introduction to the subject suitable for students is A Guide to Hazard and Operability Studies published by the Chemical Industry Association, UK[13].

Risk and Operability issues have a significant impact upon the commercial success as well as the technical success of the project. As demonstrated by GLEEP and Windscale AGR projects it is important that gated processes and staging of projects is undertaken where this leads to better control of the risk. This staging should be viewed in both the context of i-graphite retrieval, especially in very large reactors, series of similar reactors and the integration in the overall reactor decommissioning sequence.

f) Obstructed Areas / Variable Geometries

Due to internal arrangements within a reactor containment it is likely that areas may exist so that one individual tool or a configuration of that tool cannot access all areas where graphite is located. For example, on Windscale AGR direct lifting in the vertical plane was not possible in all positions due to a Corbel being present as part of the reactor pressure vessel. It is possible other fixtures and fittings may be present that require a temporary break in the retrieval process. In the case of GLEEP there was a steel tunnel that allowed irradiation experiments in the centre of the core that had to be removed before the lower bricks could be removed.

In the case of commercial reactors both spherical and cylindrical pressure vessels are common and in the case of research, demonstration and other miscellaneous reactors there are a few designs that were fuelled from the side (at least one reactor in France and Windscale Pile), or even from the bottom as is the case of Hunterston A Magnox in the UK.

The implications of obstructed access and variable geometry are: -

1) Additional tooling may be required to access the obstructed areas or variable geometries. This may require movement of i-graphite in more than one dimension. An option is to consider multiple tools working together. A proposal for Magnox has been to combine the use of hoists, beams and remote operated vehicles.
2) A break may be required in graphite retrieval production process that slows the rate of removal
3) Other materials may be removed with different chemical and radioactive characteristics that require consideration in the waste route.
4) Line of sight issues may arise and require special consideration.

g) Size reduction in-situ

In the context of figure 2 the term in-situ is taken to mean at any point in the process prior to dispatch, i.e. the size reduction process is an in line activity and like any other in-line activity it could potentially be on the critical path. The size reduction is not necessarily in the reactor containment itself; it could be but it could also be external to the reactor containment either in an extension to the reactor containment or outside the reactor containment all together (as at GLEEP). Strategically, a decision to size reduce is required either to meet integrated waste management requirements or to facilitate retrieval process requirements. The needs could be for the following reasons: -

Integrated Waste Management

1) The integrated waste management strategy requires i-graphite to be size reduced to in-feed a downstream treatment process and the best practicable means shows that it is best done in-situ.
2) The integrated waste management strategy suggests that i-graphite blocks are segregated in to low activity and high activity portions to increase disposal options. The significant variation of C-14 content within bricks was noted on the Manchester University Graphite Training Course. The concentration of C-14 is highest closest to the fuel channel.

Retrieval Process requirements

3) i-graphite blocks may be larger than the maximum dimension of the waste box immediately downstream of the point of dispatch
4) The posting position from the reactor containment is of small size compared to the smallest graphite dimension
5) Blocks may be size reduced deliberately (broken by force) to help release them from the core structure
6) Blocks may be size reduced by accident e.g. dropping.

Work Package 1 top tier retrieval options, which are concerned with the impact on the integrated waste management process, have identified bounding conditions for the retrieval and segregation process. See Table 23. + / - represents benefits disadvantages that are not reported here but may be found in a future Work Package 1 report [14]

Table 23 Bounding Options for Retrieval of i-graphite

OPTION	Particulate		Massive	
	Wet	Dry	Wet	Dry
<25 year delay	+ / -	+ / -	+ / -	+ / -
75 year delay	+ / -	+ / -	+ / -	+ / -

There are clearly advantages and disadvantages that need to be assessed jointly by the waste management team and by the retrieval and segregation team.

Focusing on the retrieval process requirements, the following issues in Table 24 need to be considered with respect to the demands of the integrated waste management need for size reduction.

Table 24 Implications of adding size reduction to the process

1.	Extra process steps
2.	Additional equipment and finding a location for its operational envelope
3.	Generation of dust
4.	Additional components to be handled
5.	Gangue materials may require sorting

Size reduction will be an extra step in the process and the impact on process timescale, cost and hazard and operability will require evaluation. The advantages of the size reduction if it leads to reduced waste volume will be significant but must be balanced against potential disadvantages to the integrity of the retrieval and segregation process.

If it is in the reactor containment, the size reduction process will require handling and systems to operate it that need to be included in the retrieval and segregation handling design; if is outside the reactor containment it will need to be integrated in to the preliminary waste route.

The implications of dust management will need to be considered. Some graphite specialists advise against reducing the size or i-graphite for this reason. It is also recommended that graphite manufacturers are consulted for their views on this issue. (Note; SGL Group and Graphtec are beneficiaries of CARBOWASTE).

Reducing the size of the i-graphite means that the number of pieces to be handled will be increased so engineering design of the size reduction process will need to devise handling systems that can manage this situation. The Work Package 1 Review Chapter for Japan notes that the opposite has been considered, i.e. to remove blocks in groups of 7 to reduce production times. GLEEP lifted blocks individually but then placed them in a basket of 16 prior to removal from the reactor containment. EdF advised that Bugey would similarly consider means of multiple removal of blocks.

The construction and monitoring design of the graphite core and the reactor decommissioning process will mean that other materials present such as pins, constraints, thermocouples, debris from decommissioning, possible lost tools, in some cases possibly even fuel could be present. Size reduction needs to take account of the presence of these materials and also to note that in some cases these materials themselves may need to be size reduced to facilitate the graphite removal.

h) Number and frequency of operations and impact of segregation

It is good engineering practice to use expensive capital equipment, tools and the work force at maximum possible utilisation. So, the minimisation of operations will encourage the minimisation of failures, faults and dose to operators. The “production engineered” process is therefore not a trivial issue. GLEEP, for example comprised 13500 blocks weighing a total of 550te and took approximately 5 months to remove all these. The ability of the contractor to optimise the process to minimise the timescale of removal was a key performance indicator. The typical commercial reactor is considerably larger and very much more radioactive. For example Magnox quantities range from 2000 to 4000te of core graphite and Ignalina RBMK has a little under 2000 te. per unit.

In considering whether to undertake an operation on i-graphite in-situ or at some other point of the integrated waste management process the local environmental condition and dose rates should be considered. For commercial reactors the dose rates in the reactor containment arise from a combination of the activation of structural components in the region, contamination that has been spread in the primary circuit (either during operation or as a result of prior decommissioning steps) and within the i-graphite itself. Once graphite has been dispatched the first two source terms are no longer relevant if the i-graphite has been segregated from the structural components, contamination and any secondary wastes (e.g. air filters) that could be packaged with it.

i) Technology Status

The failure of any equipment to function in a highly radioactive environment is a serious problem because it not only leads to cost of repair but further delays due to investigation and regulator concern. The track record of the equipment to be used, the systems needed for control and management, the ability of operators to safely operate it and the nature of the environment in which it will be operated are clearly key issues to consider.

Equipment that has been used before to retrieve i-graphite effectively and efficiently should be given serious consideration and caution expressed if equipment requires to be subject to extensive R&D and validation because it has no prior track record. That is not to rule out new and novel techniques if they bring advantages either to retrieval from the reactor containment or to integrated waste management as a whole. But failure of the reactor retrieval process will have implications far beyond the local issues and need to be carefully addressed. Programmes may be delayed and downstream facilities left idle.

Not withstanding the status of the tooling and methods, it is likely that they will require trial and practice on mock ups of the reactor containment and possibly the extended preliminary waste route. This will demonstrate the technology status and also train operators in its proper use and recovery from un-planned events. Some tooling and methods may be applicable on a generic basis, i.e. there is no variation in approach between one reactor design and another either in terms of access or expected condition of the i-graphite. But approaches to reactor designs and approaches to decommissioning are by no means generic as will be seen in sections 5.3.5 and following below. So engineers are cautioned against starting with the tool system but should follow a logical technology selection approach detailed in section 7.3. The requirements process in Table 18 and the risk reduction requirements in Table 15 should be considered.

For novel techniques, as well as development of the technique itself, extensive production engineering development will be further required to prove its reliability.

Another issue to consider with technology is its flexibility. Is it only appropriate to a specific design of core or can it be easily adapted to variations? It is unlikely that a single retrieval and segregation methodology can be found that covers Magnox, AGR, UNGG, RBMK and a host of research and demonstration reactor designs. Even a single retrieval methodology for one design such as Magnox may be difficult due to design variations within the genre. See section 5.3.6.

Section 7.3 breaks down the technology selection process in to a number of steps and once the appropriate technology has been selected its success of operation depends on the ability and training of the operator. The operator should have available a tool box of techniques to draw upon to apply to continually changing conditions within the reactor containment and adjacent areas. So the availability of a choice of flexible and reliable tooling, knowledge of the condition of the i-graphite and the reactor structure, well prepared access to the reactor containment and waste routes and prior training of the operators on simulators are the keys to technological success. The Windscale AGR i-graphite removal project is a clear example of the value of this approach. GLEEP is an example of taking that technology and making it fit for purpose on a research reactor.

Opportunity should be taken to transfer between applications at least the principle if not the actual tools themselves. For example the drill and tap method was use on both Windscale AGR and GLEEP but the support method was completely different.

j) Best Practicable Means

Caution should be used with respect to the use of this term. In the UK it has a specific interpretation by the regulators and BPEO (best practicable environmental option) and BPM (best practicable means) studies are part of the front end engineering process to arrive at a preferred solution. For retrieval and segregation it is assumed that BPEO will be relevant primarily to the Integrated Waste Management approach (covered by Work Package 1). BPM on the other hand could usefully be applied by the front end engineering team to the retrieval and segregation process design as summarised in figure 2. Both BPEO and BPM use multi-criteria decision analysis (MCDA) as a tool to facilitate the decision making process and recommendations for MCDA are a deliverable of Work Package 1. A preliminary MCDA was undertaken in the Work Package 2 workshop and the output of it is reported in section 6.

Specific national regulators should be approached to cover requirements in different member states. The following definition of BPM is taken from Scottish Environment Protection Agency (SEPA) Radioactive Glossary of Terms.

> *Operators are required to take all reasonable measures in the design and operational management of their facilities, in order to minimise discharges and disposal of radioactive waste, and to achieve a high standard of protection for the public and the environment. BPM takes account of availability, costs, operator safety and the benefits of reduced discharges and disposals.*

In meeting the above philosophy some issues that might be considered when selecting the method and tooling for i-graphite retrieval are listed in Table 25.

Table 25 Environmental Considerations for Retrieval and Segregation

1) Management of i-graphite dust i. Impact of environment e.g. air / water / other ii. Impact of size reduction of i-graphite on dust
2) Introduction of secondary wastes i. Caused by the environment ii. Caused by the tooling use iii. Caused by the tooling control, care and maintenance, and disposal
3) Waste management i. Number of waste management operations down stream (store size, transport etc) ii. Use of most appropriate waste routes (is segregation added value?) iii. Waste packaging efficiency (see below) iv. Availability of waste disposal routes (during the period of the retrieval of the i-graphite; whether <25, 50 or 75 years is an issue for Work Package 1)
4) Selection of abatement techniques and standards i. Consider the impact of operations in each step of figure 2 ii. Relate to availability of existing site infrastructure and authorisations for abatement iii. Consider abatement standards for the timing of the project

k) Waste Packaging Efficiency

Waste Packaging Efficiency can be achieved in a variety of ways of which the most obvious are firstly, integrating of waste package size and graphite size so that the minimum number of packaging containers is required and secondly, filling in gaps in the waste package or the graphite within it with smaller pieces so that voidage is minimised. The first is an engineering consideration and the second is largely down to the skill and experience of the operator. Maximum packaging that can be achieved may also be restricted by limits on the design of waste boxes for mass and radioactive inventory. That may not depend solely on the graphite but on other activated and contaminated materials mixed with it.

From the point of view of the retrieval and segregation process opportunity to influence the waste box size may not be available as, as in the case in the UK, waste box sizes have been largely standardised. Interestingly, the Windscale AGR project uses a non-standard box as the project started before waste containers were standardised in the UK.

There is nothing to stop a UK operator considering a new design of box but the process of getting the design validated is a costly constraint that must be considered carefully. There is of course the opportunity to choose from a range of box sizes. However, it is considered that choosing the box size is a matter for the Integrated Waste Management Evaluation and is therefore covered by considerations in Work Package 1. This could be a significant issue as, if shallow disposal of i-graphite is proposed as has been the case in France, then opportunities for bigger containers for i-graphite might be an option, whereas there tend to be size restrictions put on waste containers that will go in to very deep repositories.

The potential for on site disposal of some wastes might present other options for some reactors that happen to be on the same site. The point is that the preliminary waste route specification must be clearly defined so that the retrieval and segregation engineer has some basis on which to optimise waste packaging efficiency.

The issue of void avoidance is clearly within the gift of the retrieval operator and the operational planning should consider Table 26.

Table 26 Factors influencing Waste Packaging Efficiency

1. The dimensions of the waste disposal boxes
2. The dimensions of any intermediate handling baskets or furniture
3. The opportunities for segregating in to different sizes or size reducing graphite blocks at the point of retrieval
4. The opportunities for segregation graphite from non-graphite materials at the point of retrieval
5. The design of the waste route after posting from the reactor containment but prior to dispatch; this will be a location / space / and buffer storage / tooling selection issue
6. The dexterity of the operator in placing the various materials in the disposal box; this may be aided or constrained by preordained use of furniture in the boxes and by training.
7. The retrieval mode with respect to removing graphite blocks in multiples as has been proposed in Japan (groups of 7).
8. The original packing design of the reactor core.
9. The distribution of radiation between graphite blocks or other associated materials

By their nature nuclear reactor cores are designed to pack together efficiently though often with gaps left to allow for various dimensional operations through operational lifetime. So packaging calculations might like to consider this but note that blocks may become damaged during retrieval, may fall over or otherwise become disorientated. Because of the distribution of activity through the reactor core (usually the highest activity levels are found near to the centre of the core) then another useful practice for waste boxes is to load them in such a way that lower active materials are near the edge of the box. This has transport benefits. The benefits of doing this need to be looked at on a case by case basis.

Any one who has moved house will appreciate the opportunities for minimising the number of packing boxes and how the order in which components are placed in boxes and consideration of their shape and mass is very important.

For cylindrical objects with channels, two methods of waste packaging efficiency are known. Firstly, the channel may be packed with other waste. Secondly the cylinder can be split in half, third or even more so that the pieces can nest against each other more effectively in the box. The challenge is to balance the detriment of the additional cutting operations (time in a highly radioactive reactor containment and time and dose in a low activity situation done manually) against the cost saving of the smaller number of boxes (and any BPM benefits). As previously mentioned the impact on dust burden of cutting needs to be considered.

l) Grout in Situ

This might be considered an alternative to retrieval and segregation. It has not been considered in any detail in Work Package 2 except to note the approach for which the reactor pressure vessel is filled with lightweight grout followed either by storage in situ or the lifting of the reactor pressure vessel and removal from its operational location to another storage location. The latter approach has been adopted at Julich for the AVR with storage on the same site (see section 5.3.9). In-situ storage using a grout of bentonite / cement has been noted for some reactors in Russia, reported in the Work Package 1 Review Report [7].

5.3.5 Consideration of Reactor Design

Retrieval and dispatch of graphite cannot be achieved unless access to and from the reactor containment has been achieved and, when that has been achieved, not until other sub assemblies within the reactor containment that block access to the i-graphite have been removed. So, a fundamental step is to understanding the reactor design, how it was constructed, how it might be modified, what containment it offers and the different uses and locations of i-graphite.

In many reactor dismantling cases it is possible that the access route may be dictated by other reactor dismantling considerations and the graphite may not be the critical activity in choosing the best access route. This may place a constraint on the choice of methodology for the i-graphite. There may be some reactors, GLEEP was an example, where this is not necessarily the case and retrieval of i-graphite may be the controlling activity.

There are likely to be a number of alternative access routes in to the reactor containment, which will need to accommodate all the process activities shown in figures 1 or 2 together with people and materials as listed in Table 8. The routes considered or used to date have already been listed in Table 9.

The route in to the reactor containment may be influenced by what is available, what already exists, whether a reactor pressure vessel is cylindrical, spherical, or cuboid and specific designs of containments such as that for the RBMK, which does not have a pressure vessel. Appendix 9 included a paper by Poskas and Poskas of LEI, written especially for Work Package 2 that covers lessons learned from core replacement at Leningrad NPP RBMK.

In the case of GLEEP, a cuboid graphite pile, which had a bioshield but no pressure vessel, access routes changed with time; initially access was from the top for both materials and men but later a side entry route also became available for man access as the level of graphite was reduced below a side tunnel that entered the core of the reactor and had been used for experiments.

The UK Magnox and the French UNGG come in two basic civil structures, viz. steel pressure vessels, which tend to be spherical, and concrete pressure vessels, which tend to be cylindrical. Research and demonstration reactors come in various arrangements some cubical, some cylindrical.

In order to understand the access and other challenges for the key legacy commercial reactor designs, three presentations were given at the Work Package 2 workshop. These described the Magnox Hinkley Point A and Oldbury reactors (used as reference designs), the UNGG at Bugey and Ignalina RBMK, The full power point presentations are given in Appendix 3 and the detail included is summarised here together with some additional information on a number of German reactors including the AVR and THTR demonstration reactors. The reader is reminded that additional information about the various reactor designs may be found in the review report for each country reported by Work Package 1.

For the Magnox designs Hinkley Point A is considered a representative steel pressure vessel design and Oldbury a representative concrete pressure vessel design. There are other differences even in these two generic types and across the Magnox Fleet – see Appendix 3.1 Magnox presentation. For example cores are pinned by Zirconium (Hinkley) or keyed by Graphite (Oldbury). Sometimes graphite tile spacers are used. One reactor, Wylfa contains an anti-lifting device to prevent lifting of graphite.

As for the UNGG, Bugey represents a concrete pressure vessel of which there are 4 such reactors in France. The other 2 have steel pressure vessels. Oxidation of graphite in Bugey was very high and 44% weight loss has been reported.

The RBMK design is restricted to Ignalina in the EU. However, globally, the RBMK designs in Russia and Ukraine, for example are expected to be of similar principle but of different scale. RBMK does not have a pressure vessel as it is a channel reactor. There is, never the less, a containment vessel through which the channels pass. Unlike the Magnox and UNGG designs which operated in an atmosphere of Carbon Dioxide, the RBMK operated in an environment of Nitrogen/Helium intending to reduce oxidation of the graphite.

It should be noted for those interested in previous experience GLEEP and Windscale AGR reactors and their graphite removal were described at a joint Work Package 1 and Work Package 2 workshop on 18th February 2009 at Manchester. These presentations are reported separately in Work Package 1.

Slide 15 of the Magnox presentation below includes a comparison of core sizes and orientation for a range of reactors. This is repeated as Figure 6.

Another important issue with respect to the placing of the preliminary waste route and services is the position of the reactor with respect to the building in which it is located. An obvious place to locate the waste route might be on the charge face and so it will be necessary to review the status of this area with respect to the removal or reuse of equipment on the face and how access is gained to the reactor containment.

For Windscale AGR where space in the charge area was very limited it was a challenge to find location for the preliminary waste route and the maintenance cell for the retrieval process. The initial solution was to jack up two of the four heat exchangers that were located in their own bioshields very close to the reactor bioshield and create space underneath these heat exchangers. As it turned out the heat exchangers were fully removed prior to the retrieval work commenced so with hindsight the jack up process could have been eliminated. This emphasises the point that a well integrated decommissioning programme will understand when, and in what order, all operations, including i-graphite removal will take place; this understanding will help to ensure that infrastructure refurbishment, removal, modification or replacement will benefit the integrated programme.

The pile cap on Windscale AGR was removed first to gain access to the reactor dome, which was removed before the dismantling machine was placed on the reactor pressure vessel and the contents of the reactor were removed in 9 work packages. Graphite from the core was only removed as the 6th campaign. Some non core graphite was removed earlier as part of the Neutron Shield dismantling (5th campaign). Other graphite in the thermal column and reflectors was removed at different points in the programme (10th campaign) when access was most convenient.

Presentations on three major reactor designs to the Work Package 2 workshop are outlined in sections 5.3.6 Magnox, 5.3.7 UNGG and 5.3.8 RMBK. In addition to give some indication of variations that can occur with HTR and test reactor a presentation on German reactors is included in section 5.3.9. (N.B. Core illustrations diagrammatic not representative.)

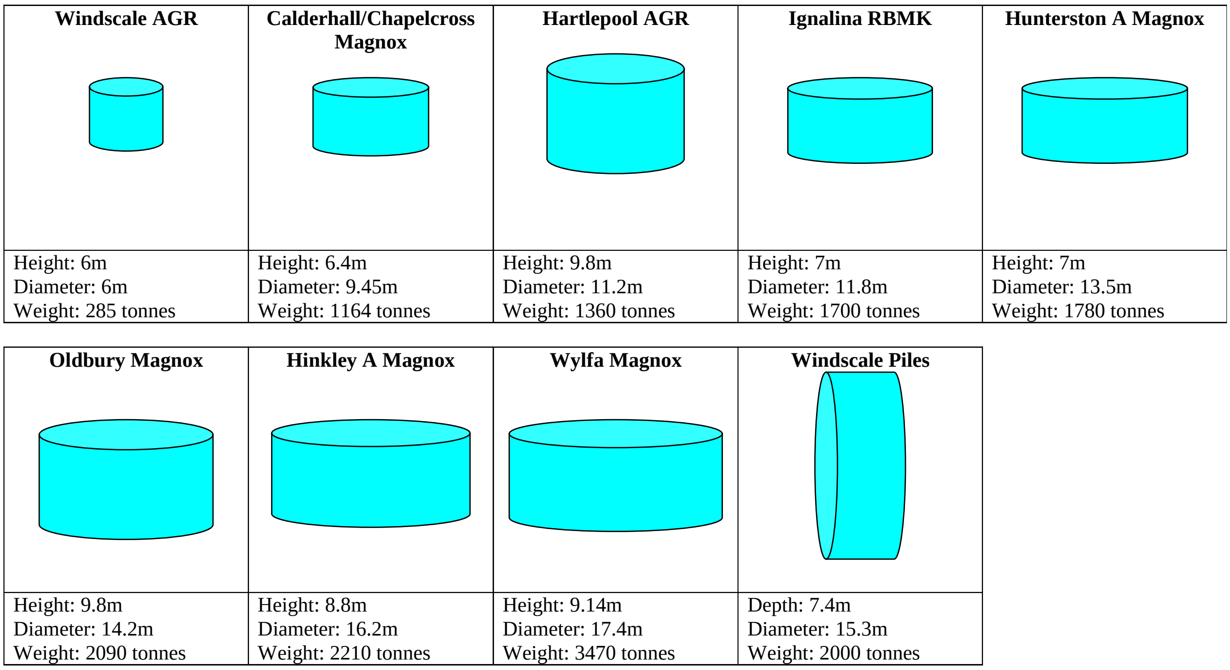

Figure 6 Various Core Geometries

5.3.6 Magnox

Michael Grave, Doosan Babcock presented an overview of the Magnox design including the core layout, brick design, restraint system, keying, presence of instrumentation, separation tiles between layers. See Appendix 3.1. Discussion raised the possibility of reverse engineering the construction method to dismantle the core. The point was raised as to whether the graphite components should be removed as found, or should be size reduced within the reactor containment prior to removal. The latter option presents the opportunity to utilise existing small penetrations into the reactor containment. Access to the core through existing large gas ducting in earlier Magnox designs was also discussed. The range of spherical steel and cylindrical concrete pressure vessels was noted that will impact upon access.

It was noted that Hinkley Point A (figure 7) and Oldbury (figure 8), whilst being typical of the range of the range of designs, did not fully represent all possible variations. For example, Wylfa has anti-lift restraints that prevent the top layer of graphite from moving vertically during fuel extraction procedures - see Appendix 8. Other variations such as the steel specification for the reactor pressure vessel evolved with time. A unique variation on the Magnox design is the bottom fuelled reactors at Hunterston A.

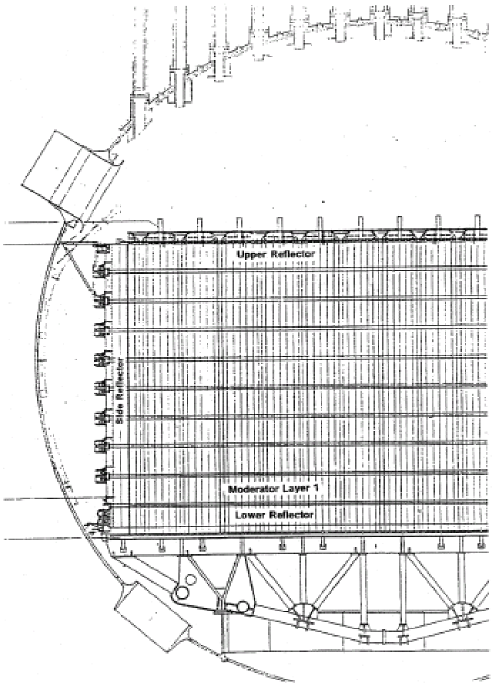

Figure 7 Hinkley Point A core layout

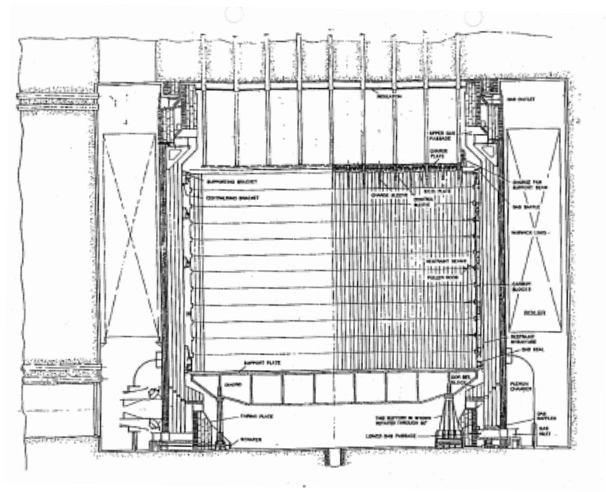

Figure 8 Oldbury core layout

5.3.7 UNGG

Francis Berenger, EdF gave an overview of the design of the prototype Bugey 1 (Figure 9) nuclear power plant including the core layout, brick design and restraint system and brick locating system (dovetail design). See Appendix 3.2. The presentation also included the proposed underwater core dismantling order and the proposed graphite waste route. Light diffraction through the water was discussed but it was explained that dismantling activities would be viewed vertically so diffraction of light does not become an issue. The need for water clarity was discussed and it was noted that the water clarity would be maintained through filtration.

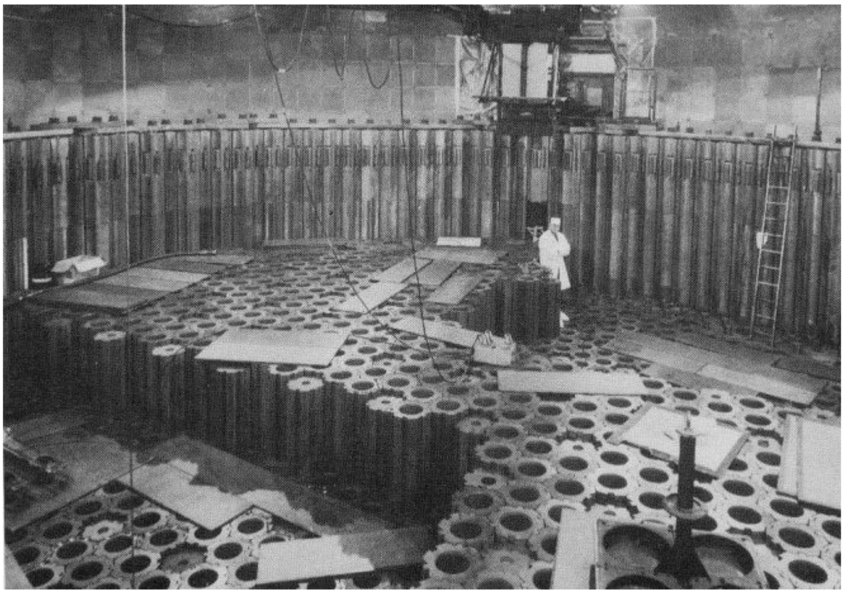

Figure 9 The Bugey Reactor Core under construction

Weight loss is a major issue with this particular reactor, with up to 40% weight loss measured in certain areas of the reactor.

In addition, merging of blocks due to expansion will need to be overcome and studies have been performed to consider appropriate release of the core restraints and the identification of specific 'key' blocks that once released (through brute force) will free the majority of its neighbours and hence the whole graphite layer.

Similarities between the proposed Bugey dismantling technique and decommissioning activities at Fort St. Vrain nuclear power plant in the U.S were discussed and it was suggested that EdF obtain further information on the keying system and brick dimensions used at Fort St. Vrain.

The added value of the use of water to control dust, the handling of graphite totally underwater up to the point of placing in a lifting container, the changing water level to close to the graphite and the ability to seal all reactor penetrations was noted.

5.3.8 RBMK

Povilas Poskas, LEI gave an overview of the RBMK at Ignalina (Figure 10). See Appendix 3.3. The presentation discussed the core layout, top level metal structure and reactor hall layout. Differences between the RBMK and other graphite moderated designs were highlighted, see Table 27 particularly the key distinction that the RBMKs do not have a pressure vessel as the coolant pressure is contained within tubes in the core channels.

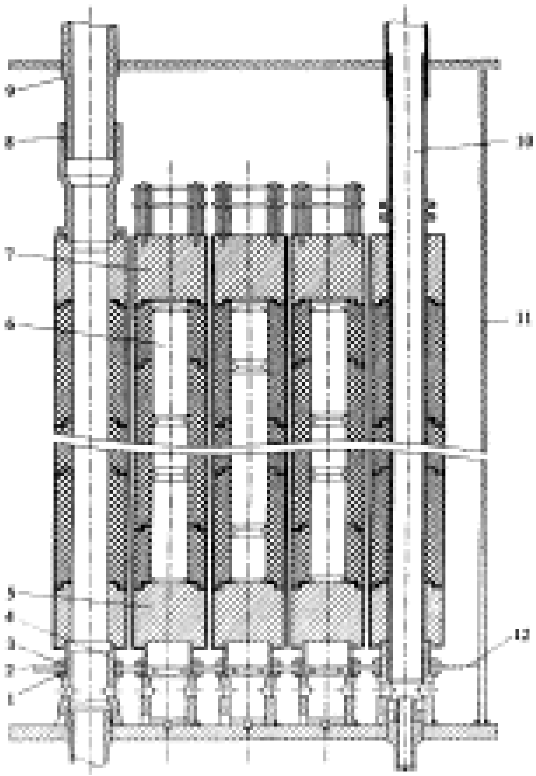

Fig. 4.3 Segment of the graphite stack
1 - top ring, 2 - diaphragm, 3 - bottom ring, 4 - bushing, 5 - steel support plates, 6 - graphite rods, 7 - shield plates, 8 - standpipe, 9 - guide pipe, 10 - reinforcing tube (reflector cooling channel), 11 - outer steel shell, 12 - ring plate

Figure 10 Layout of the Ignalina Core

A description was provided of the unique removal of a column of graphite bricks at a matching plant in Leningrad. See Appendix 4. The Leningrad NPP Unit 3, work was carried out in an outage and not during a decommissioning campaign. A passage used for steam-gas exhaust was installed at the top of the bioshield of the reactor; it was of small diameter, typically 570mm minimum. The columns of graphite bricks that were removed through this passage were not close to it in their reactor core position but could be removed because channel replacement work was in progress at the same outage. Consequently a route was devised for moving damaged graphite masonry along a path where fuel channels had been removed until they were in a position below the steam-gas exhaust passage. Thereafter 250x250x600 and columns of 15 blocks were lifted out of the reactor through the passage. The possibility of using this approach for decommissioning of the Ignalina reactors was discussed after the presentation.

Table 27 Key Differences Between RBMK and other Gas Cooled Reactors

Feature	Ignalina RBMKs	Typical Gas Cooled Reactor
Long channel components	Fuel and cooling water channels, control rods etc, below standpipes	Details vary, but basically channels for fuel, cooling gas and control rods below standpipes
Large area at charge face	Access to channels is through removable shield blocks at vault floor	Access to channels is through removable shield blocks at charge face
Above core structures	Top cover, steam/water Pipework bundles	Details vary, but basically, guide tubes, neutron shield, and pipe bundles
Graphite stack	Graphite moderator and reflector comprising several thousand blocks	Graphite moderator and reflector comprising several thousand blocks
Core support structure	Core support structure for graphite stack and other components	Core support structure for graphite stack and other components
Cylindrical structures	Diaphragm, water cylinder and sand cylinder around graphite	Core restraint structure around graphite. Not a continuous structure, but still requiring 360 degree access
Pressure Vessel	Doesn't exists, pressure is contained in the channels, the major key design difference, may or may not affect dismantling, according to option chosen etc.	Spherical and cylindrical pressure vessels, either steel or prestressed concrete
Concrete structures	Reactor is surrounded by the concrete vault	Reactor is surrounded by the concrete bioshield, either separately in the case of steel vessels, or integrally for prestressed concrete vessels
Penetrations	Large openings through the concrete vault for steam and water pipes, standpipes for fuel, control, and instrumentation	Penetrations through the bioshield for gas cooling, inlet and outlet, standpipes for fuel, control instrumentation

5.3.9 German Reactors

Werner von Lensa, FZJ gave an overview of decommissioning activities in Germany and noted that 2 helium -cooled high temperature pebble bed reactors have been built and operated in Germany that require retrieval of graphite from their cores. The AVR reactor has a small pressure vessel and so the decision was made to fill its interior with grout and remove as a single unit. The THTR used two different grades of carbon material in its construction, the main core is a pure carbon while the reflector is a less pure carbon-stone. This carbon-stone layer has 100x more radioactivity associated with it compared to the purer graphite. Both reactors utilised Helium coolant so comparisons with other reactors may be appropriate. Dust was considered to be a major issue, especially with respect to the carbon-stone. There is also a limited amount of graphite (reflectors, thermal columns) with low activation from water-moderated research and material test reactors in Germany.

The AVR is a 15 MWe experimental reactor at Juelich and was shut down in 1988 after 21 years of operation. The 300 MWe prototype Thorium High Temperature Reactor THTR was operated from 1985 until 1989 in Hamm-Uentrop. They are at different stages of decommissioning.

The moderator graphite is an integral part of the spherical fuel elements of AVR and THTR. Until the end of the seventies a closed fuel cycle concept with reprocessing of the thorium-based spent fuel - similar to the U.S. - was developed in Germany. It included mechanical disintegration of the fuel elements, incineration of the moderator graphite, and recovery of the fissile material from the fission products by extraction. In early 1980 the Government decided to stop R&D on reprocessing mainly due to economical reasons. Since then, the back-end option for the spent fuel is based upon direct disposal in a deep repository without reprocessing, thus using the special safety features of the fuel elements, i.e. the coated particle fuel stabilised in a graphite matrix.

The spent fuel elements have been removed from both reactors.

Graphite and low-purity carbon were used in both reactors for reflector, thermal insulation and shielding purposes, amounting to 225 te. (AVR) of ceramic materials in total and about 550 te. plus 52 te. irradiated graphite pebbles and about 10 te. absorber elements, in the case of the THTR. A radiochemical characterisation of the AVR ceramic materials was performed, leading e.g. to carbon-14 levels between 100 kBq/g for graphite and 9 MBq/g in maximum for low-purity carbon.

A 10 MW research reactor (Merlin type) with two thermal columns was operated at Juelich for 23 years and decommissioned to greenfield. 11 te.of irradiated graphite with a medium carbon-14 activity of 71 Bq/g are being stored at Juelich waiting for disposal.

A 23 MW heavy-water material test reactor (DIDO) was operated at Juelich for 45 years and shut down in 2008. The irradiated graphite from reflector and thermal columns amounts to 30 te. Radiological analyses are in preparation.

5.3.10 Generic Reactor Design Features

The previous four sections illustrate the considerable differences between reactor types and even within specific reactor types in terms of core design, general reactor containment configuration and the local infrastructure so that, whilst a generic process for retrieval and segregation has been defined (figure 2), a generic design for a reactor is clearly not readily achievable.

With in common reactor designs there are differences that have been noted in this report within the Magnox fleet (noting there are two outside the UK in Italy and Japan that have some similarity to either Berkeley or Bradwell) and with in the UNGG fleet in France. The EdF decommissioning planning for Bugey and the UNGG fleet in general has highlighted that at least two retrieval and segregation approaches will be used for its 6 UNGG reactors; four that have concrete reactor pressure vessels will use the under-water approach and two with steel pressure vessels will be decommissioned in-air. In the UK the Magnox fleet, which is mainly steel pressure vessels also includes two concrete pressure vessel stations at Wylfa and Oldbury. Within the steel pressure vessel class there are variations in the specification of the pressure vessel steel that may lead to variations in the local dose rates caused by steel activation. (e.g. Calder Hall). Lower dose rates could affect the choice of manual/semi-remote/fully remote or shielded by water approaches. The order in which stations in a fleet are decommissioned could also be affected if dose levels are a key consideration in timing, not withstanding ownership and funding issues.

It is noted that the final stations of the Magnox fleet are starting to show features at least in layout that are similar to the following generation of AGR plant. For example the heat exchangers are contained inside the reactor bioshield in an annulus, which is a potential route for access to the core.

Features that are important to i-graphite retrieval and segregation that have been observed as a result of Work Package 2 are listed in Table 28.

Table 28 – Variable Generic Reactor Features of Relevance to i-graphite Retrieval

	Feature	**Options / Notes**
1	The shape of the reactor containment	Spherical, cylindrical, cuboid
2	The orientation of the reactor containment and its fuel charging system	Vertical access from top generally, a few horizontal, some charged from below
3	The height to diameter ratio of the core	Affects reach, internal movements and internal space for operations
4	The position of the core within the reactor containment relative to other reactor plant	Such as thermal, neutron shields, heat exchangers
5	The various uses of graphite within the reactor containment,	E.g. core moderator, reflectors and use in thermal shields and thermal columns
6	Internal obstructions to the graphite	Structural, operational, or accidental
7	The volume and tonnage to be removed	Affects economic tooling selection and scheduling of the waste route
8	The construction material of the pressure vessel	Affects level of activation and hence background radiation dose
9	The geometry of the core and its components	Different block sizes, and shapes, tiles, pins etc.
10	How the core is stabilised, supported and constrained	See 9, consider as found stability, e.g. graphite morphology and weight loss.
11	How the core is monitored	Presence of thermocouples is commonly found
12	The area of the charge face or other local working areas	Preliminary waste routes, services, personnel and tool routes need space
13	The status of the containment	Impacts on containment and structural

		integrity
14	The potential to interfere or impact with adjacent plant	Safety and Scheduling
15	The potential to interfere or impact with a building in which the reactor is located	Small research reactors are often located in older buildings (e.g. GLEEP) Rare cases of operation in preserved historic buildings
16	Other items	See Table 27. Fleet operators should investigate reactor variations (e.g Wylfa anti-lift devices

5.3.11 Retrieval of stored graphite in unconditioned state

The presence of irradiated graphite in and unconditioned / unpackaged state has been reported at a number of sites in France (all UNGG operational graphite waste stored at St Laurent, Marcoule and La Hague), from Magnox at Berkeley and Hunterston in UK and Tokai Mura in Japan, at Vandellos in Spain and from RMBK reactors. The reader is recommended to read the Work Package 1 review report country chapters[7] for a more complete picture.

At the time of writing the only known European completed retrieval project for this type of waste that has been reported in detail is the project to remove i-graphite from silos at Vandellos in Spain. The waste is packaged and stored on the reactor site. In the UK similar activity is required at Berkeley and Hunterston A and these projects are in various states of planning or completion and will progress in accordance with Nuclear Decommissioning Authority priorities. The waste is know as Fuel Element Debris and has other reactor components mixed with the graphite. Outside Europe Work Package 1 Review Report indicates that fuel element sleeves have been incinerated at Tokai Mura implying that the i-graphite waste has been retrieved. No details are presently available.

As part of its contribution to the CARBOWASTE WP2, ENRESA, Spain has produced a report on the Vandellos project. This document explains comprehensively the challenges, methods and results of the project and is therefore presented in full in Appendix 5.

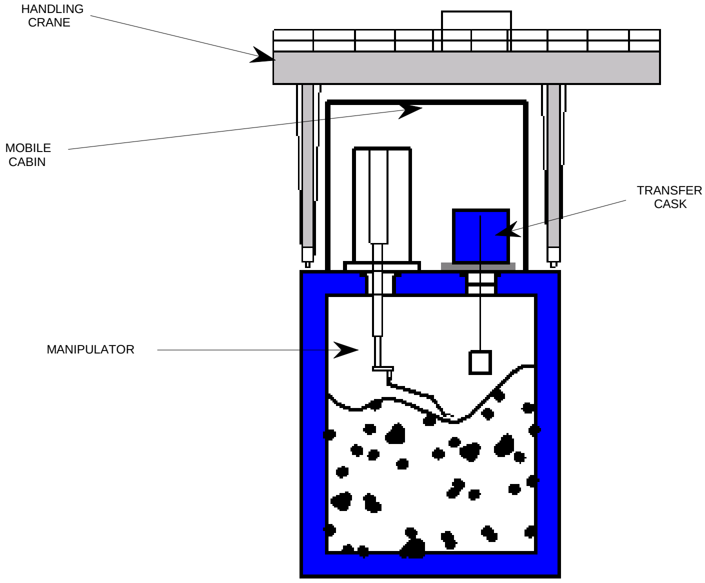

Figure 11 Vandellos Waste Selection and Retrieval

The information in the Vandellos report is summarised in Table 29

Table 29 Selected information form Vandellos Report

Information or Operation	Generic interest to Retrieval and Segregation
Silo Geometry	1500 m3 capacity 25.50 x 8.70 x 10.25m deep
Waste Characteristics	Graphite Sleeves and Cylinders plus other materials and 2 spent fuel elements with graphite in 1 silo. See Appendix 5 for quantities and source terms
Waste Management Process Flow Diagram	Refer to Appendix 5 figure 3.1.1
Retrieval and Transfer System Functions	1. pre selection of graphite and retrieval 2. transfer to conditioning building 3. equipment layout see figure 11 below
Tools used	1. Petal grab type tool able to grab and load bags, steel pieces and drums. 2. Spade type tool for pushing and loading 3. Rake tool for rearranging, pulling and loading.
Sorting and conditioning systems	Sorting may be of interest to Segregation see Appendix 5 for more detail. Included a graphite shredding machine. (Note Graphite was also shredded on the GLEEP project)
Process data for graphite treatment	Approximately 70 sleeves per hour 10-35mm size after shredding. Compare 50mm used for GLEEP
List of Auxiliary Systems	1. Ventilation system 2. Fire protection system 3. Electrical system 4. Fluid system 5. Radiation Monitoring system 6. Monitoring system 7. Alarm system
Programme	Refer to Appendix 5. Total concept to completion – 4 years Silo 3 contained only graphite sleeve; it was emptied in 6 months. 1054te graphite packaged.
Operational lesson 1	Retrieval manipulator failure but project could continue because of redundancy of tools – petal grab used.
Operational lesson 2	Dose rates in graphite boxes higher than expected because of failure to fully separate steel wire.

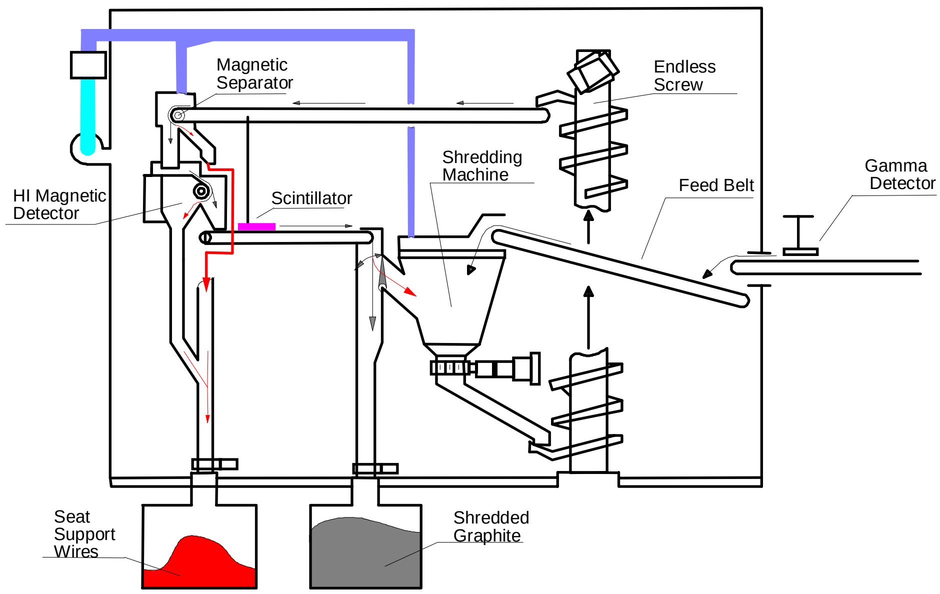

Figure 12 Vandellos Graphite Conditioning

5.3.12 Draw upon previous experience

The retrieval and segregation of i-graphite process can be divided in to the areas in Table 30 for which the previous experience is listed. The headings correspond with the retrieval and segregation process as defined in figure 2.

Table 30 Graphite Retrieval and Segregation Process Experience

Table Notes:
1. **Country** in bold see WP1 Review report[7].
2. Information underlined is included in this report or an Appendix to it.
3. The WP2 team members experience covers: graphite reactor planning / safety case / cost studies/ implementation, retrieval of unconditioned graphite from stores, tooling selection for sampling and retrieval, integrated waste management solutions and modelling, graphite material science, reactor configuration.

Activity	General experience	Graphite Specific reported in CARBOWASTE
Planning	This document includes checklists that apply to many decommissioning situations including graphite. Reference to the general conference literature and IAEA documents is recommended.	1. **France** The EdF approach to Bugey 2. **UK** GLEEP and reference [15] also some additional information herein 3. **UK** Windscale AGR also some additional information herein 4. **USA** Fort St Vrain – WP1 but only limited information published 5. **Japan** – retrieval proposals for Tokai Mura 6. Modelling – will be a future output of WP2 – initial requirements defined
Sampling	Not covered in detail as it is an existing practice in UK for AGR reactors and in France for UNGG	1. Information from Samples on CARBOWASTE has been defined in WP1 and results will be provided by WP3. 2. **France** also EdF presentations at EPRI Graphite workshops
Access to the Reactor Containment	This document includes checklists that apply to many decommissioning situations including graphite. Reference to the general conference literature is recommended	1. Leningrad NPP core replacement 2. Magnox presentation 3. UNGG presentation 4. RBMK presentation 5. GLEEP and **UK** 6. Windscale AGR and **UK** 7. Vandellos Silos – see Appendix 5 8. **USA** Fort St Vrain

Table 30 continued

Handling / Removal	This document includes checklists that apply to many decommissioning situations including graphite. Reference to the general conference literature and IAEA documents is recommended.	1. Leningrad NPP core replacement 2. GLEEP see planning 3. Windscale see planning 4. Vandellos Silos – see Appendix 5 5. Fort St Vrain see Planning
Segregation	This document includes checklists that apply to many decommissioning situations including graphite.	1. GLEEP see planning also some additional information herein 2. Vandellos Silos – see Appendix 5
Dispatch	This document includes checklists that apply to many decommissioning situations including graphite.	1. GLEEP see planning also some additional information herein 2. Vandellos Silos – see Appendix 5 3. Windscale see planning also some additional information herein

Further information on experience of graphite retrieval and segregation is given in Section 7.

5.3.13 Checklists and design process guidance

In planning a i-graphite retrieval and segregation project it is useful to have checklists so that managers, engineers, regulator and stakeholders can be confident that all relevant issues have been taken in to account. This report has been designed to include a number of tables that in many cases form checklists. These are summarised in Table 31.

Table 31 Checklists Summary

Subject	Reference	Section
STRATEGIC ISSUES		
Contributors to this report	Table 1 List of Participants	3.2
Good Practice	Fundamental Prerequisites and Good Practice	4.2.1
State specific requirements	Policy and Regulatory Approach	4.2.2
Issues not covered	Exclusions	4.2.3
Strategic issues	Bounding conditions / engineering prerequisites	4.2.4
Activities to support i-graphite retrieval	Table 2 Associate processes and Operations	4.2.4
ENGINEERING ISSUES		
Operations on i-graphite	Table 3 i-Graphite Handling	4.2.4

	Operations	
Useful data from sampling	Table 4 Graphite and Materials Property Information	4.2.4
Flow diagram (graphite piece)	Figure 1	4.3.1
Engineering Solution steps	Table 6 Engineering Hierarchy for Graphite Retrieval Solution	4.3.3
Waste specification	Table 7 Preliminary Waste Route Constraints	4.3.4
Access needs	Table 8 Functions of Reactor Containment Access	4.3.5
Access Routes	Table 9 Potential Access Routes	4.3.5
Graphite forms and components	Table 10 Reactor Graphite and Associated Components	4.3.6
Supporting facilities	Table 11 Ancillary processes necessary to support Graphite Retrieval	4.3.7
GENERIC RETRIEVAL PROCESS		
Plan Access to Core Retrieve / Handle / Segregate / Dispatch	Figure 2 represents the generic process and the key requirements for each of these stages are listed following for each stage of the process.	4.3.8
Sampling	Sampling has already been covered in section Table 4	4.2.4
PROJECT SOLUTION ENGINEERING PROCESS		
CARBOWASTE Toolbox Hierarchy	Figure 3 – Hierarchy of the Retrieval & Segregation Toolboxes	5.1
Integrated Waste Route	Figure 4 – Irradiated Graphite Route Map – level 1	5.2.1
Integrated Waste Route Step 2 Retrieval and Segregation	Figure 5 - Retrieval of i-graphite and/or other options	5.2.2
Criteria for Retrieval and Segregation upper level used in WP1.	Table 12 Upper level criteria for retrieval and segregation	5.2.3
Key sub-processes for retrieval and segregation	Table 13 The key sub process of the Retrieval and Segregation process	5.2.4
Retrieval and Segregation parameters	Appendix 7 Groups and Factors influencing Retrieval and Segregation Criteria	5.2.4 & Appendix 7
Production Engineering elements	Table 14 Retrieval and Segregation Production Line Components	5.2.5
Choice restrictions imposed by the Preliminary Waste Route	Table 15 Preliminary Waste Route Constraints on Retrieval Process	5.2.5
Operations - Upper Level Risks	Table 16 Upper Level Approach to	5.3.2

	Reducing Operational Risk	
Planning Parameters (refer to Fig. 2)	Table 17 Planning Parameters	5.3.2
Project Specific Retrieval and Segregation Specification	Table 18 Developing the Generic Retrieval Process for Specific Applications	5.3.2
Retrieval and Segregation Staged Approach	Table 19 Gated Process for Developing the Retrieval and Segregation Process	5.3.2
Handling/Removal, Segregation, Dispatch Parameters (Refer to Fig. 2)	Table 20 Handling/Removal, Segregation and Dispatch Parameters	5.3.3
Sampling	Table 21 Requirements for Sampling	5.3.4 b
Major Hazards	Table 22 Potential Major Hazards	5.3.4 e
Specific Hazards and operability issues	Appendix 9 Hazard Identification and Operability Check list	Appendix 9 and 5.3.4 g
Upper Level Criteria	Table 23 Bounding Options for Retrieval of i-graphite	5.3.4.g
Size Reduction implication	Table 24 Implications of adding size reduction to the process	5.3.4 g
Best Practicable Means	Table 25 Environmental Considerations for Retrieval and Segregation	5.3.4 j
Waste Packaging Efficiency	Table 26 Factors influencing Waste Packaging Efficiency	5.3.4 k
Reactor design variations	Table 27 Key Differences Between RBMK and other Gas Cooled Reactors	5.3.8
Reactor Design Feature variations	Table 28 – Variable Generic Reactor Features of Relevance to i-graphite Retrieval	5.3.10
Retrieval from Vaults and Silos	Table 29 Selected information form Vandellos Report	5.3.11
Retrieval Experience	Table 30 Graphite Retrieval and Segregation Process Experience	5.3.12

5.4 Solution Engineering Process

Figure 3 in section 5.1 summarises the process in pictorial terms.

5.4.1 Principles and Upper Level Design

The upper level steps for retrieval and segregation of i-graphite is as follows.

Understand the Gated Process

The gated process is summarised in Table 19. It will be applied to the total decommissioning project for the system in which the i-graphite is found and to sub-projects with in it. It is necessary to know how the i-graphite project will integrate into this process at each stage, i.e. planning, safety case and implementation. Different organisations and groups will be involved at each of these stages and their role and responsibility also needs to be clearly understood.

This report has assumed that the planning stage of the gated process will commence with actions associated with the integrated waste route (as defined by Work Package 1) to define the preliminary waste route for the i-graphite before the sub-project to retrieve and segregate i-graphite from the reactor containment continues to its own planning, safety case and implementation activities, which are covered by Work Package 2 recommendations.

It should be clear, however, that this decision is arbitrary and project integration is equally important to waste management integration. Local variations due to specific regulator requirements should be recognised and individual policies of i-graphite owners may impose other constraints.

Examine the requirement of the Integrated Waste Route.

Reference will be make to the Route Map Figure 4, section 5.2.1, which will be adapted to recognise the strategic approach to i-graphite waste management.

Examine the information available and required for retrieval and segregation

Reference will be made to the Step 2 of the Route Map Figure 5, section 5.2.2, and as far as possible a programme of work established to gather the necessary data to design the retrieval and segregation activity.

Define the upper level constraints of the Preliminary Waste Route

The integrated waste management study will identify the outputs for retrieval and segregation based on the outputs evaluated against the criteria in Table 12 and will have considered bounding conditions for i-graphite retrieval as signified in Table 23. This will lead to the constraints on the preliminary waste route as defined in Table 15.

Outputs from Upper Level considerations

The above considerations are likely to be subject to multi-criteria decision analysis as will be described by Work Package 1. These will recommend preferred options but the onus will remain on the retrieval and segregation engineer to define the best practicable means and some iteration with the integrated waste route engineers is almost certain to be necessary.

Additional constraints imposed by System Decommissioning

There may be additional constraints on the preliminary waste route imposed by other reactor decommissioning activities and these should be added to the Preliminary Waste Route specification.

At this point the design process for retrieval and segregation can proceed in more detail

5.4.2 Project Engineering process

Basic system constraints

These should be examined as summarised in Table 18. What is the retrieval and segregation end point. This will largely be the defined preliminary waste route requirements supplemented by organisational and infrastructure constraints.

Develop the flow diagram and preliminary design work breakdown structure

It will now be possible to take the generic flow diagram in figure 2, taking account of the operations on the i-graphite in figure 1 and generate the specific flow diagram (or perhaps some optional variations on it). In developing this diagram it will be important to ensure that the key issues in Table 16 are addressed, viz. how is contamination trapped, how is radiation protection facilitated and how is good visibility of the process achieved. Major hazards as outlined in Table 22 will have been addressed and the outline safety systems to compliment Figure 2 identified.

Optioneering and Development of Concept Design

It is next desirable that the preferred engineering options are derived for each of the process steps in figure 2 and checklists to consider are found in Tables 17, 20 and 21. This will ensure that alternative processes or methods are compared consistently against the criteria, which will be discussed in section 6.

Design Work Breakdown Structure

Once the preferred approach(es) for i-graphite retrieval and segregation has been agreed the work breakdown structure for the project and each of the key stages (planning, safety case and implementation) as indicated in Table 19 can be finalised. Figure 2 will be the basis of the work breakdown structure, which contains at least the elements in Table 32.

Site Operational Engineering

Item 1.7 in Table 32 is a very important aspect of the process and should not be left to the end. The site engineering will vary substantially from case to case. The GLEEP and Windscale AGR represent extremes in-air, the former allowing man-access and the latter the need for fully remote operations. Never the less it is those engineers who have experience of reactor installation and construction, through life maintenance and decommissioning who will at the end of the day make the designed process a success or not. It will be they who have the greatest experience of the constraints of working on a nuclear site, of using operations that are considered to be safe, reliable and flexible, who can appreciate how to do things quicker and simpler. It is strongly recommended that such engineers and reactor operators are involved in design teams at the earliest possible time in the process. Such engineers have been consulted by Doosan Babcock whilst compiling this report.

Implementation

It is going in to depth and detail beyond the scope of CARBOWASTE to discuss this issue in detail. However, the information in this report can be used in the contractor prequalification stage for i-graphite to identify contractor attributes that are likely to lead to successful bids; at the tendering stage to ensure that complete tender specifications can be prepared and risks that contractors cannot easily manage are identified. At this and the contractual stage the report can be used to check that the work breakdown structure is complete and that responsible persons have been identified for the project, sub-projects and tasks. Also it should be clear that the i-graphite retrieval and segregation process is integrated with the following project plans

1. The site decommissioning plan
2. The integrated waste management strategy for i-graphite
3. The reactor dismantling plan

Table 32 Model Work Breakdown Structure for Retrieval and Segregation

Ref	Task	Supporting Checklists
1.0	**Project Management and process integration**	Tables 17,18,19,29,31*
1.1	Integrated waste management	Tables 6,7,12,18,23 Figs 3,4,5
1.2	Retrieval and segregation process engineering	Tables 2,6,10,11,13,18, Figs 1,2
1.3	Health, Safety and Environment BPM/ALARA and waste management	Tables 5,7,10,15,16,25
1.4	Safety cases and regulatory affairs	Refer to each WBS task
1.5	Development (prior experience, technique proving, training, test rigs, mobilisation,)	Refer to each WBS task for gaps in knowledge and 1.4 requirements
1.6	Risk management	Tables 6,7,16,22
1.7	Site operation engineering	Tables 6,7,8,14,15,16,19
2.0	**Data Acquisition**	Table 30
2.1	Plant data records	Table 4
2.2	Sampling	Tables 3, 4, 21
2.3	Modelling	Table 4
2.4	Traceability (to support waste management)	Tables 4, 7
3.0	**Access to the Core**	Table 28
3.1	Site Infrastructure	Tables 2, 6, 18
3.2	Creating Access	Tables 2, 6, 8, 9, 18
3.3	Access to the i-graphite	Table 18
3.4	Working environment (air, water, other)	Tables 8, 11, 18
3.5	Structural assessment	As for 1.0 and 3.0
4.0	**Handling / Removal**	Tables 20, 28
4.1	Define i-graphite and material condition	Tables 6, 10
4.2	Core structure management	Tables 6, 10
4.3	Visualisation and monitoring	Table 10
4.4	Tool selection	Tables 3, 7, 8
5.0	**Segregation**	
5.1	Size Reduction if required	Tables 3, 7, 24, 26
5.2	Waste stream separation	Tables 3, 10, 26
5.3	Other processes as required	Tables 3, 7, 26
5.4	Integration with 3.4	See 3.4
6.0	**Preliminary waste route**	
6.1	Specify route	Tables 4, 7
6.1.1	Downstream constraints	Table 7
6.1.2	Up stream requirements	Tables 5, 7
7.0	**Other site/reactor specific requirements**	Table 29 (silos and vaults)

These are the prime checklists for each work breakdown structure item. The reader is advised to use all the reference data in Table 31 as is appropriate.

6.0 Criteria and Characteristics of the Process Options

6.1 Define the criteria for selection for retrieval and segregation

6.1.1 Integrated waste process

Reference has already been made to the task 1.6 in Work Package 2 that led to the criteria for evaluation of retrieval and segregation of i-graphite as part of the integrated waste management process. These criteria have been presented in Table 12 and discussed in section 5.2.3.

In essence the evaluation of the upper level criteria leads to the definition of the Preliminary Waste Route, which is the starting point for the retrieval and segregation engineer.

6.1.2 Specific to retrieval and segregation issues

The criteria for evaluation for the design of the retrieval and segregation process are sometimes the same as the upper level criteria but in general are different.

The retrieval and segregation engineer should start from the constraint of the Preliminary Waste Route and then engineer the best solution that will work for the particular reactor. It has been recommended already that early involvement in the upper level planning is desirable so that the Preliminary Waste Route is as flexible as possible to give the retrieval and segregation engineer maximum choice.

It is not within the scope of CARBOWASTE Work Package 2 to study specific retrieval and segregation criteria but the list in Appendix 7 and already referred to in section 5.2.3 represents a starting point for engineers developing a list of criteria.

The opportunity was taken at the workshop, to undertake a trial MCDA for retrieval and segregation process. This analysis is at the boundary of Upper level and Lower level analysis as it followed the consideration of retrieval and segregation issues and experience by the Work Package 2 team. The analysis is presented in section 6.1.3 and 6.1.4.

6.2 Application of MCDA to Potential Retrieval and Segregation Processes

6.2.1 Defining the process

A presentation by David Ross of NNL, who is leading the CARBOWASTE Work Package 1 Multi-criteria Decision Analysis may be found in Attachment 12 of the Work Package 2 Workshop minutes [16]. It outlines the MCDA process and provides three illustrations that show generic process flow diagrams for , retrieval and segregation, treatment and disposal. Examples of strategies for each of these is presented.

The focus of the analysis in the workshop was on retrieval and segregation but in the context of the integrated waste management solution. Four typical integrated waste management solutions were identified that were thought to represent many of the situations likely to be encountered in practice. The following elements were identified: -

1. The retrieval medium (choice air, water, or air / inert)
2. The delay <25, 50 or 75 years
3. In-situ treatment (none, chemical or size reduction)
4. Disposal conditioning (grout or gasification)
5. Disposal (deep, intermediate, shallow or capture of the C-14)
6. Dose impact on operator (low, low / medium or high)

Theoretically, over 600 combinations are possible from the above elements, but four based on current strategies in vogue were selected and are presented in Table 33.

The first represented what became referred to as the “UK approach” because of the long delay and current base case is deep disposal, the second the “French approach” because it mirrors the strategy for Bugey. Other options chosen were in-air in less than 25 years, an approach that might be necessary for those who wish to decommission at an early date but who have reactor designs unsuitable for filling with water. A fourth recognised the proposals to capture C-14 by gasification with a 50 year delay being chosen for completeness rather than for specific reasons.

Table 33 Integrated Waste Management options for i-Graphite

Option	Retrieval Medium	Delay	In-Situ Treatment	Disposal Conditioning	Disposal	Operator at Workface
1	Air	75 Years	No treatment	Condition (grout)	Deep disposal	Low dose rate
2	Water	<25 Years	In-situ chemical treatment	Condition (grout)	'Shallow' disposal	Low dose rate
3	Air	<25 Years	None	Grout	Intermediate Depth Disposal	High dose rate
4	Air/Inert	50 Years	Size reduction	Gasification	Capture	Low/medium dose rate

Clearly, others might come up with other combinations. The reason for undertaking this exercise is more to demonstrate the process for MCDA than to select a specific option.

6.2.2 Selecting Criteria of Assessment

The key headings for criteria following the review in Appendix 7 are:

1) Political/Social/Economic
2) Timing
3) EHS&Q
4) Waste Management
5) Reactor Containment / Core Design
6) Security

Ignoring Reactor Containment and Design that can only be looked at in specific circumstances, the criteria were organised in the form of key words to use to challenge the processes listed in section 6.2.1. Following a detailed review of the Upper level and lower level criteria, David Ross recommended grouping the key words in to the following areas: -

1) Environmental and Public Safety (this relates to the Best Practicable Means)
2) Worker Safety (this relates to the working environment encountered)
3) Security (this relates to management of nuclear materials)
4) Technology (this relates to “will it actually work?”)

6.2.3 Examination of two current process options

The results of the workshop analysis for the UK and the French options are presented in Tables 34 and 35 respectively. The break down of the challenging keywords is found in the left hand column and the workshop team identified key issues by group discussion. The combined workshop then identified control measures that would mitigate adverse effects. Comments in italics have been added for clarification.

6.2.4 Discussion of the MCDA

It is stressed that this is a demonstration of the MCDA approach. The focus was on retrieval and segregation not on comparing UK versus France. Work Package 1 MCDA will examine the broader issues. The analysis did confirm the importance of the factors in Table 16 i.e.

- How to trap contamination
- How to facilitate radiation protection
- How to ensure good visibility of the work zone

and the importance of balancing dose reduction against loss of knowledge and infrastructure for in-air due to delay and balancing the great benefits of under-water to reduce dose and dust against the need to manage the water for clarity and containment. Both processes are feasible.

Table 34 Option 1 – UK

Keyword	Issues	Control Measures
Environmental and Public Safety - Radiological - Man	Lower dose situation Lower impact to the environment, people	*Appropriate to 75 year status*
Environmental and Public Safety - Radiological - Environment	Low significant effluent No potential of leak 75 years reduces impact Dust/aerosol more of an issue than dose	Negative pressure ventilation system Filtration i.e. in the containment building, Building maintenance/ refurbishment required
Environmental and Public Safety - Resource Usage	No water use No interim storage required Infrastructure may have to be rebuilt Future cost unknown	None significant
Environmental and Public Safety - Non-Radiological		*Normal civil engineering procedures*
Environmental and Public Safety - Nuisance (to people and environment)	Rebuilding infrastructure Transport	Normal civil engineering procedures
Environmental and Public Safety - Solid Wastes	Filters No resins as for water	
Environmental and Public Safety - Catastrophic Accidents	Loss of containment less of an issue than with water Lifting Structure degradation Weather Other unknowns (e.g. demographic)	Monitoring of gases Fire extinguishers Building maintenance/ refurbishment/ Surveillance
Worker Safety - Radiological	Lower dose rate but possible airborne contamination More semi 'hands on' operations possible	Normal civil engineering procedures Temporary shielding
Worker Safety - Non-radiological		Normal civil engineering procedures: 'best practice'
Security		Maintain proper physical protection
Technology - Maturity	Proven at WAGR	
Technology - Operational	Loss of services Decrease of knowledge	Good design

Table 35 Option 2 France

Keyword	Issues	Control Measures
Environmental and Public Safety - Radiological - Man	Dust and gas issues are a lower risk No accidental discharge?	
Environmental and Public Safety - Radiological - Environment	Resin disposal (same disposal route) Reactor sealing required Still need to ventilate system Water loss issue • Contamination • Loss of shielding	Secondary containment required Evaporation dealt with Water treatment required Discharge authorisations Environmental control e.g. temperature
Environmental and Public Safety - Resource Usage	No major structural works No interim store required	
Environmental and Public Safety - Non-Radiological	Chemical additives to water e.g. pH	Control of water additives: acid (controls algae growth)
Environmental and Public Safety - Nuisance (to people and environment)	Not as much as option 1 Disturbance after only 25 years instead of 75 years	
Environmental and Public Safety - Solid Wastes	Resins	Good design + operational control
Environmental and Public Safety - Catastrophic Accidents	Catastrophic water loss, Dropping from top of reactor (very high)	Secondary containment Seismic substantiation Substantiated lifting procedures
Worker Safety - Radiological	Water shielding External dose from steelwork Internal dose during tool removal	Washdown/contamination procedures Secondary shielding and removal of steelwork as work progresses
Worker Safety - Non-radiological	Drowning	Normal pond procedures + design
Security	Higher radioactive inventory	*Graphite is maintained in an enclosed system from underwater to encapsulation*
Technology - Maturity	Previously proven (Fort St. Vrain and other pond work)	
Technology - Operational	Tools suitable for underwater use Visibility? Only suitable for certain reactor designs	Adequate design of tools Needs careful control of particulate control

6.3 Identification of other critical issues for decision making

Taking account of the presentations, discussion of the parameters that input the retrieval and segregation process (Figure 2), the MCDA criteria and the application to generic designs, the workshop team were asked to maintain a record of comments and issues as the workshop progressed. These are recorded in detail in the workshop report[16] but some of the more important observations are presented in Table 36.

Table 36 Observations at Workshop

	Environmental and Public Safety
1	Dust management requires resolution (but note: both GLEEP and Windscale AGR achieved i-graphite retrieval despite dust, particularly in the case of Windscale AGR)
2	Utilities needed for management of the working environment
3	Exploitation of the existing infrastructure
4	Waste packaging efficiency brings clear benefits
	Worker Safety
1	Physical mock ups are very important
2	Multiple access routes to the reactor should not be overlooked
3	Different safety issues for manual and remote operations
	Technology selection
1	Reactor design and operational records
2	Question the constructors of the reactor
3	Consider the impact of the packing factor in the reactor pressure vessel (note EdF found that high packing favoured use of water and low packing did not)
4	To design the handling / removal system start with the tool that interfaces with the graphite
5	Multiple access routes to the reactor should not be overlooked
6	The core status viz. jammed or loose blocks requires consideration
7	Exploit the use of standpipes for retrieval (note the RBMK experience)
8	Research reactors of relatively simple geometry and lower dose should use fit for purpose solutions but can learn from this report
9	Safestore may impact on access considerations including loss of infrastructure
10	Versatility and flexibility of approach essential
11	Very important to take account of background experience and information
	MCDA and Supporting processes
1	Computer models will require correct input information or be ineffective
2	The challenge of developing a scoring system for MCDA

3	Knowledge management and reduction of uncertainties is the best stage approach
4	MCDA will help extract the general components of the process
5	Preparation of presentations to the format of the generic process (figs 1 or 2) helps in initiating the problem analysis
	Workshop and Project improvements
1	Terminology glossaries for members from different national backgrounds
2	Try and segregate high level and low level criteria (note the series of workshops in Work Package 1 and 2 that have considered retrieval and segregation have tried to gain a clear understanding of the definition of the Preliminary Waste Route that is crucial end-point for retrieval and segregation. The two cannot be totally separated. This report has endeavoured to make clear what are high level issues (mainly post the Preliminary Waste Route) and what are low level issues (mainly prior to dispatch of the i-graphite from the reactor containment)
3	Collaboration on modelling between EdF and NNL may be beneficial
4	An IAEA TecDoc for retrieval and segregation might be useful based on the CARBOWASTE Work Package 2 output.

7.0 Information and Tools

7.1 Summary of Retrieval and Segregation experience

7.1.1 Reactors

The prior experience on retrieval of i-graphite from nuclear reactors has been summarised in Table 30 in Section 5.3.12. The key reactor systems are:

GLEEP UK undertaken in-air and of relatively low activity so that man access was permissible. The project is summarised in the UK report of the Work Package 1 Review Report [7]. For additional information a paper by Grave et al, presented to the IBC Decommissioning Seminar is reference[15] and a presentation to the Work Package 1 / 2 workshop in Manchester in February 2009. The prime lifting mechanism for the graphite blocks was drill and tap, placing a group of blocks in a basket, hoisting them out of the reactor and lowering to a floor monitoring station and shredding plant prior to dispatch in plastic drums for thermal treatment. Outside the reactor blocks were handled by vacuum lift and conveyor table.

Windscale AGR UK undertaken in air but remotely as the background radiation level was too high for man access. The project is summarised in the UK report of the Work Package 1 Review Report [7]. For additional information a presentation to the Work Package 1 / 2 Workshop in Manchester in February 2009. The Windscale AGR was decommissioned in nine phases, not all involving the removal of i-graphite. The predominant i-graphite phase included the removal of the neutron shield campaign 5 and the removal of the core in campaign 6. Access to the reactor was by removing the head of the pressure vessel and installing a remote dismantling machine, which supported various manipulator arms and hoists. The tooling to remove the i-graphite was supported from both arms and hoists and tools used were ball race for blocks with holes in them, drill and tap, grabbing, sweeping and vacuum.

Fort St Vrain was undertaken underwater and is covered in the USA report of the Work Package 1 Review Report[7]. Published information is primarily about the decommissioning project as a whole and any reference to graphite suggests that a defueling route used in when the reactor was in operation was used and that it also went to the same operational waste use. So the project is primarily of interest in terms of demonstrating the reactor access underwater. A very similar approach is being proposed for Bugey in France and, whilst this project has not yet been implemented, reference may also be made to the France chapter in the Work Package 1 Review Report[7] and to Appendix 3.2 of this report.

Leningrad NPP has been described in this report following presentation by LEI at the workshop on possible approaches at Ignalina NPP. This method which involved moving fuel channels out of the way and creating an access route to the reactor containment through a steam release penetration is described in Appendix 4 of this report and summarised in section 5.3.8.

There is very little other specific mention of i-graphite retrieval in the literature but much of the i-graphite retrieval process is common to overall graphite moderated nuclear reactor decommissioning. Apart from Windscale AGR none has been decommissioned to date but it is known that planning studies have been done for Magnox NPPs in the UK, Italy and Japan.

For example a paper by G. Holt entitled *Radioactive Graphite Management at UK Nuclear Power Stations* [17] concluded that

> *"the actual techniques, e.g. for cutting and materials handling, used in dismantling the reactors will remain the same regardless of the time when the dismantling is undertaken. Generally the only variable will be the degree to which remote operations will need to be used to apply the techniques. With respect to reactor graphite, the cores are made up of a stack of graphite blocks and dismantling will be the reverse of the original construction, e.g. removal block by block in a sequenced manner. The blocks have holes passing through them that form the fuel channels. These holes can be used during dismantling to allow the insertion of a mandrel for lifting purposes".*

In the case of Japan the study reviewed lifting graphite blocks in groups of 7.

7.1.2 Vaults and Silos

POCO takes place at an early stage in the decommissioning of a nuclear reactor and this may require retrieval and segregation of i-graphite arising as a result of through life nuclear power plan operation.

Such retrieval and segregation has been demonstrated at Vandellos I, Spain and is a requirement at Magnox power stations at Berkeley and Hunterston, UK and for RBMK such as Ignalina, Lithuania and many plant in Russia. Whilst not completed in the UK either planning or projects have been underway for several years to consider retrieval of wastes from vaults.

This operational waste is, in most cases, the result of the accumulation of fuel element debris, such as graphite struts and sleeves. The application of retrieval and segregation from a debris store is different from the application to a reactor core. The variation will include factors such as: -

1) geometry and access
2) local environmental conditions,
3) the size and mix of materials to be retrieved
4) the likely activity levels to be encountered both in the store and in the graphite
5) the quantity and quality of materials to be handled.

Never the less there are many issues in common such as the need for remote operations, retrieval and separation of materials and the graphite material nature. Lessons may therefore be learned that could usefully be reviewed before embarking on the design of a reactor core retrieval and segregation project.

A report covering the experience at Vandellos has been reviewed in section 5.3.11 and is included in full in Appendix 5. Additional comments are made in section 5.3.11 concerning the partially completed project at Berkeley UK and the plans to empty vaults at Hunterston A.

7.2 Recognised strategies

Reviewing the key reactor designs of Magnox, UNGG and RBMK, their proposals for decommissioning, the approach for the AVR in Germany and the prior experience on GLEEP and Windscale AGR has enabled a number of strategic approaches to be identified. The strategy selection needs to consider a hierarchical approach.

Level 1 Interfacing with the Preliminary Waste Route
Level 2 Providing the necessary segregation
Level 3 Working environment to manage contamination, dose and visibility
Level 4 Access to the Reactor (but see note).
Level 5 Define specific tooling

Note: It is not straight forward to allocate a hierarchical position for *access to the reactor*. In strategic terms it is a variable and needs to be considered in the context of total reactor decommissioning. In defining access preferences in studies specific to i-graphite retrieval needs it is recommended that the best access is defined after the requirements of level 1, 2 and 3 have been defined. Clearly potential routes need to be identified earlier to facilitated level 1 - 3 decision analysis. In the case where total reactor decommissioning has been considered and a access route chosen for reactor dismantling, then *access to the reactor* should be considered a constraint but that does not preclude consideration being given to secondary access routes.

Levels 1, 2, 3 are further discussed below in section 7.2 and defining specific tooling is considered in section 7.3.

Preliminary waste route

All strategies require that the Preliminary Waste Route is defined before proceeding to design the retrieval and segregation process.

Baseline approach

Waste dispatched from the preliminary waste route will be grouted and disposed of in standard disposal boxes.

As long as graphite blocks have no dimension larger than the proposed waste disposal box size reduction is not required.

As long as all wastes primary and secondary can be placed in the same waste disposal box and go to the same waste disposal route, no segregation is required.

Sorting of waste is desirable for waste packaging efficiency purposes.

Radiation segregation may be required if (probably non-graphite components) there is risk that the radioactive capacity of the disposal box might be exceeded for transport purposes.

Waste boxes may alternatively be filled in a waste sorting facility that is external to the reactor pressure vessel or may be in an extension to the reactor pressure vessel containment.

This baseline case may be carried out either in-air or under-water.

If in-air there is no impact on the waste disposal by the environment except that HEPA filters may be added to the waste stream that are required to collect any i-graphite dust generated in the process.

If under-water, the waste source term may be modified because mobile radioisotopes may be removed by the water and any dust generated removed in water filters.

The Zero Option

The AVR at Jeulich has been subject to what might be termed a “Zero Option”. In this case the i-graphite has not be retrieved but instead the reactor pressure vessel has been filled with light-weight grout and the whole reactor pressure vessel will be lifted out of its current building and transported to a new facility on site for interim storage. The solutions beyond that time are not yet defined. Should the i-graphite (and some fuel is also present) require retrieving then segregation or not from the concrete grout will be a challenge.

In most commercial reactor cases, the removal of the RPV is not considered a likely option and it is recognised that AVR is not representative of current legacy reactor designs. Those considering specific research reactor designs may consider unique solutions if it is appropriate to do so. However, in Russia grouting and leaving the reactor in place has been practiced.

Non baseline strategies

If the base line strategy for the preliminary waste route is not followed then alternative strategies for Segregation need to be considered; these are described in section 7.2.2.

7.2.2 Segregation

Segregation Strategies

The options for segregation are:

1) Graphite and its associated gangue materials and secondary wastes are segregated in to intermediate, low level and in some cases very low level waste or free-release streams after retrieval but prior to dispatch.

2) Graphite is segregated from all other non-graphite materials and transferred to graphite only waste boxes. These may be further segregated as for 1) above. This may be done at the core location or elsewhere within the extended reactor containment.

3) Graphite is treated in-situ in the reactor containment to remove or reduce radioisotopic contamination (H-3, C-14, Cl-36). Then either 1) or 2) above are followed.

4) Graphite is selectively retrieved from within the reactor containment in accordance with its perceived radioactive condition. Then either 1) or 2) are followed.

5) Graphite is size reduced in situ to facilitate its removal and handling from the reactor containment or as a requirement to meet the size distribution requirements of a downstream process or in order to split blocks in to high and low activity sections. Size reduction may be either to a smaller bulk but still recognisable graphite blocks or to a fine state (small pieces or powder). Only bulk size reduction can seriously be considered for activity level segregation.

6) A waste packaging and treatment facility may be constructed adjacent to or in the reactor containment and operated in conjunction with any of steps 1) to 5). If it is a process operating in-line with retrieval / handling it is considered part of the retrieval and segregation integrated process prior to dispatch. If it is located off-line, usually at a location on site but remote from the reactor it is not considered to be part of the retrieval and segregation process.

Segregation activities at point of retrieval

1) The definition of the 6 strategies and variations above does not imply that they are all feasible or desirable. Whilst benefits may accrue to the Integrated Waste Management process, significant difficulties may arise in application inside a reactor containment. Processes may become easier to apply in delayed decommissioning scenarios when activity levels have significantly decayed but in that case some of the reasons for segregation may have disappeared. The following paragraphs introduce some of the considerations that influence feasibility.

2) It may be necessary to segregate non-graphite materials such as pins, thermocouple, asbestos insulation if present from the graphite. There is probably space in most reactors for separate waste baskets or hoists to be used to remove materials separately; some local cutting or segmentation may be required. Alternatively, miscellaneous bits and pieces can be left in position to fall to the bottom of the reactor to be retrieved later. This is the most likely scenario as it has been demonstrated as feasible at Windscale AGR but was not practiced deliberately. If these pieces are subsequently separated in to intermediate, low level or other waste streams this could theoretically be done in-situ but is more practicable in a dedicated waste sorting area as used at Windscale AGR. An issue is confirmation that a waste is say LLW rather than ILW. This can be more readily done in a low background radiation area. If this is not the case, then sampling of each individual block may be required unless a suitable assured model is available. Sampling can add considerable time and cost to the retrieval and segregation process and may require samples to be sent off-site with a resulting time delay in obtaining results.

3) With respect to graphite only waste streams, the level at which this is achieved should be clearly understood. Graphite is certainly going to be associated with radioisotopic contamination but other substances may be present due to materials carried around the circuit in operation or due to decommissioning procedures associated both with graphite removal and not. In the case of under-water it has been noted that the graphite will be wetted and the isotopic contamination status is likely to be changed.

4) With respect to in-situ treatment, this could be considered incidental or deliberate. An example of incidental treatment is the case of decommissioning underwater in which case it is known that Cl-36 levels in the graphite are expected to reduce (reported by EdF). It is planned that Cl-36 levels can be safely managed by external ion-exchange treatment thus avoiding build up in the water. There has been no report of using this method to deliberately decontaminate the graphite in-situ or by other chemical treatment methods to enhance recovery of any radioisotope. There is still considerable debate about the mechanism of contamination of the graphite going on in the CARBOWASTE project and beyond and this, together with the complex chemical engineering that would need to be introduced in to a difficult environment, in most cases, suggest that in-situ treatment is not considered feasible on the basis of current knowledge.

Equally there have been no reports of in-situ gas-phase treatment. For the same chemical engineering reasons it is felt that if gas-phase treatment such as steam reforming is undertaken to decontaminate graphite, this is best done off-reactor. Work packages 4 and 5 are investigating treatment and recovery processes and no information proposing in-reactor treatment has been put forward to date. Proposals to size reduce graphite to grain size, 1-50mm have been put to EPRI and were reported at EPRI graphite sessions in Lyon in 2008 and in Hamburg in October 2009. Both processes are at very early stages of conception.

5) Selective removal of i-graphite blocks from regions of the reactor could be considered so that account is taken of variations due to the reactor flux being smaller in outer regions of the core and highest in the centre. There are also other i-graphite pieces present in, for example thermal shields (as Windscale AGR) that was not segregated by process but segregated by phasing of the project.

These wastes may also fall in to different activity streams because of their function in the reactor. Tooling may need to be changed for different i-graphite blocks because there purpose is different. Core or neutron shield graphite may contain vertical holes to allow fuel elements or control rods to pass through, reflector graphite on the perimeter may not have such holes.

The strategy to deal with all the different forms of i-graphite will need to balance the advantage of segregation versus the impact on the programme for retrieval due to additional operations. Also the more complicated the process the greater the risk of breakdown and need for maintenance and potential in some cases for increased dose to operator or need for more remote maintenance cell equipment. Separate removal of thermal shield graphite and core graphite by programme management was practiced at Windscale AGR. The issue of deciding, which blocks fall in to which radioactive category would have to be done either by modelling or by sampling, with the consequences to operations already discussed above.

6) Specific graphite blocks might be split in to high and low activity sections. The activity in a the region of a graphite block close to a fuel element hole is known to be higher than for a position away from the hole. Trepanning tooling could be envisaged that could cut around the high active area with the hole in it perhaps using similar tooling that created the hole in the first place. This is considered an unlikely scenario as it would add very significantly to the time of graphite retrieval and general advice is to avoid graphite cutting operations wherever possible.

7) On the basis of current technology status, it is judged that segregation in the reactor can be justified if this aids removal of the blocks from the reactor containment (if this accelerates the process), reduces hazardous situations or risks to the retrieval tooling and handling system, allows the graphite to pass size related restrictions and benefits waste packaging efficiency. There needs to be space in the reactor containment to undertake the activity and the relative costs of segregating in situ and segregating in an external facility needs to be understood.

8) Segregation may also be achieved incidentally by phasing the reactor dismantling activity as was the case at Windscale AGR.

Segregation activities at a waste sorting cell external to the reactor pressure vessel

1) Apart from the justified in-situ segregation further opportunity exists to segregate if a waste sorting cell is built in to the retrieval and segregation process close to the reactor pressure vessel and in an extended reactor containment prior to dispatch. Such was the case at Windscale AGR. The sorting cell was used to facilitate radiation monitoring of the recovered i-graphite (and other materials retrieved from the reactor at different phases) and for operations such as weighing, facilitating waste packaging efficiency or to assist correct waste packaging for transport purposes.

   Other options for location of the sorting cell within the extended reactor containment might be in heat exchanger locations , which in some cases (AGR in general and to some extent Magnox) are accessible via a large penetrations or gas ducts.

2) Waste dispatched after appropriate quality checks is likely to be placed in a waste disposal box or a transfer box, if it is going for further treatment. The placement of the lid on the box and grouting is not in the scope of this study. But for information waste dispatched from Windscale AGR waste was immediately grouted after leaving the sorting cell and, with an allowance for curing buffer storage, could be considered an in-line activity. Proposals reported for Windscale Piles, where there is a Wigner Energy issue, have suggested that the graphite be boxed, interim stored and grouted at a later date.

3) To facilitate the efficient operation of the sorting cell and the retrieval activity care needs to be taken to understand which of these two activities is on the critical path and where possible capacity or buffer storage must be optimised if graphite is to be removed to programme and cost. It was found on GLEEP that graphite was removed from the reactor considerably faster than the capacity of the shredding machine. In this case, temporary packing of the graphite and buffer storage was used and the shredding operation continued for some time after graphite removal had been completed. Increasing the shredding capacity by duplicating machines would have been an alternative. Each case needs to be considered on economic merits, space available and other downstream operations.

4) The waste sorting sell can be located outside the reactor containment but fairly close to it. An appropriate location might be on the reactor charge face if sufficient room is available or in an adjacent turbine hall or service areas provided these facilities have previously been decommissioned. Ignalina for example has a very large charge face and the reactor building is abutted on to the turbine hall. This option requires a suitable handling system and containment to move the i-graphite from the reactor pressure vessel to the sorting cell. For low energy reactors as exemplified by GLEEP an area on a floor of the building in which the reactor is located can be allocated for such activity.

5) The waste sorting cell may be in a waste treatment and packaging facility located elsewhere on the nuclear site and designed to accommodate a multitude of wastes form different decommissioning projects on the site. In this case a transport system is

required. In the context of CARBOWASTE this is out with the scope as the production process can now have buffer capacity built in so that the rate of removal from the reactor containment and the rate of i-graphite segregation are not on a critical path.

6) Much of the above discussion centres around i-graphite retrieval in-air. The presentation on Bugey decommissioning in Appendix 3 describes the planned approach underwater. In this case the status in paragraph 4) above is being followed. Waste is posted from the reactor containment in to a transfer box that is then lifted by crane in the reactor building before being lowered into a transfer tunnel, where it is transported into the adjacent turbine hall. Still in the tunnel it is grouted, checked and then dispatched.

7.2.3 Working Environment

Based on the analysis by EdF and summarised in Table 16 the three key operational risks in the retrieval of i-graphite are the spread of contamination, failure to protect the workforce and process hold up due to difficulty in seeing or interacting with the workface. This last point is not immediately obvious until it is considered that due to graphite condition, geometry, reactor construction and distance a highly variable activity is taking place that is likely to require considerable operator decision making at the time of operation. A key challenge of the i-graphite retrieval and segregation process design will be the need to avoid killing flexibility and choice. Retrieval and segregation is not like a normal production line process where a constant product is being manufactured. The graphite blocks may well have been very similar when installed, their final condition may be subject to prediction by modelling, yet still the retrieval process will encounter the unexpected. Good line of sight to the work piece is fundamental to success as these variations need to be understood in real time.

The strategic choice is controlled by three conditions.

1) Choice of delay for decommissioning
2) The detail design of the reactor
3) Choice of tooling operational mode, e.g.

    a. Fully remotely controlled systems
    b. Mixed manual / mechanised systems
    c. Manual based systems operated with shielding protection

Delay period for decommissioning

In general <25 year delay will require that operators cannot get close to the work face due to high background dose rates from activated components in the reactor especially if Co60 is present in the steel (e.g. stainless steels) , less so if steels used did not contain cobalt. All sources will need to be considered including contamination.

There may be cases with some research reactors where, for early decommissioning, dose needs to be taken in to account but is not a major constraint as far as dismantling goes.(It may be a constraint for waste disposal options and may, because of that, impact upon Preliminary Waste Route (refer to GLEEP).

For 75 year decommissioning, significant reduction of dose is to be expected and the opportunity for man access is improved. It may not be appropriate to use full man access for three reasons, doses must be as low as reasonably achievable, conventional accidents can be reduced by limiting man-access and production throughput considerations may require the process to be mechanised in any case. (This latter point was the case at GLEEP).

For 50 year decommissioning an intermediate position is to be expected but the requirements need to be considered on a case by case basis.

The 50 and 75 year cases are more likely to suffer from infrastructure loss than the <25 year case but that does not mean the even 25 year infrastructure is fit for purpose; either because the plant design life is past or because its specification is more appropriate to reactor operation rather than decommissioning or i-graphite removal.

So the most important issue is to understand the radioactive inventory and background working conditions at the planned timing of decommissioning.

Detailed design of the reactor

The study by EdF for choice of environment for Bugey is the reference case that has been considered by Work Package 2 and has been discussed in this report. There strategy was founded on previous demonstration of feasibility in the decommissioning of Fort St Vrain in the USA. Bugey (UNGG) and Fort St Vrain (HTR) are very different; the access to the reactor internals is very similar, i.e. remove reactor head, and install a lowerable platform upon which men can work, fill the reactor pressure vessel with water, which acts as a shield and lower the platform and the water level as the reactor dismantling proceeds from the top down.

EdF noted however, that the under-water approach was not suitable for all reactors in the UNGG genre. It is considered appropriate for the concrete pressure vessel reactors but not for their steel pressure vessel reactors. So Bugey and 3 others that have concrete pressure vessels are expected to be decommissioned as early as possible whilst the 2 steel pressure vessel reactors will be decommissioned later in-air. The difference in the design is more the reason than the fact that the pressure vessel is concrete or steel material. For example the presentation on Magnox in Appendix 3.1 shows the difference between the cylindrical concrete pressure vessel of Oldbury versus the spherical pressure vessel of Hinkley Point A as well as core design differences.

It is known that under-water decommissioning was considered by UKAEA for considering the decommissioning of Windscale Pile 1 (the fire damaged reactor). This was not a normal situation and the choice of working environment needed to take account of the damage and unusual radioactive and chemical conditions within the reactor. Never the less for that particular reactor sealing all penetrations was an issue (it has a vertical charge face) and the same problem arose for gas sealing (argon) as well. EdF have indicated that some sealing will be necessary in Bugey both for penetrations and for the heat exchangers that are located below the reactor compartment. But that in the case of the Bugey reactor sealing is feasible.

EdF also noted that the choice between under-water and in-air was influenced by the packing of components in the reactor. For a very high packing density, the decision came out in favour of water and for low density of packing in favour of air.

Slide 5 of the UNGG presentation in Appendix 3.2 refers to a Scenario Matrix that EdF used in coming to the decision between using air and water starting from the view point of a high dose rate area and the decision impacted upon the choice of tooling.

Working under-water

It was noted that the i-graphite was not critical in the choice between in-air and under-water, the global decommissioning operation was of greater influence on the choice and in any case many of the tools used for global decommissioning will be utilised or adapted for i-graphite. The comparison between a transparent shielding protection in the case of water and the need for remote robotic tools supported by remote visual equipment for in-air was a key consideration to remove the risk of not being able to see or interface easily with the work face. The advantages of water in providing both radiological protection and contamination control (dust not an issue) meant that on the one hand manual type long reach tools operated by workers on a platform above the water and on the other complex remote engineering was not necessary. An air filtration system with capability for a high dust burden is not required but on the other hand water treatment to removed suspended solids and maintain water condition and clarity is.

To be able to use these simple tools the strategy is: -

1) to operated directly above the work face so that there is no diffraction of light issues
2) to lower the platform and water level so that the operator is as close to the work face as possible and has direct line of site

It is not considered that operating under water should be a specific constraint on the availability of tooling. Underwater operations are common place in many industries including nuclear and some environments e.g. on oil platforms located in the sea are considerably more arduous.

For further discussion of the may issues of choice between in-air and under-water retrieval the workshop MCDA demonstration reviewed the in the proposed UK and French approaches and this is reported in section 6.2.4 and Tables 34 and 35.

A tooling selection strategy has been developed by AMEC for this project and is presented in section 7.3 below.

7.2.4 Access

Access possibilities in to and out of the reactor have been identified as a primary constraint on the retrieval options available. This subject has been discussed throughout this report but particular attention is drawn to sections 4.3.5, 4.3.8 and 5.3.5.

7.3 Technology Selection: Tools and Techniques

Arriving at the most suitable tooling and techniques to retrieve the i-graphite from the reactor and to segregate it requires a process involving options, challenges and constraints. A logical and staged approach to the final solution is required and a flow sheet approach recommended. The following discussion is an analysis of a flow chart produced to guide the process, which is presented in figure 13 with each operation in the flow chart cross-referenced to a relevant section of this report. This section has been added to the first draft of the report as it was produced by AMEC (Jon Montgomery with Michael Grave support) as an action from the workshop.

7.3.1 Define Retrieval Challenge

In addressing the challenge of retrieving i-graphite, the engineer needs to know: -

1) What is the condition of the i-graphite to be handled?
2) Where is the i-graphite?
3) How can it be reached?
4) How can it be handled (closely influenced by answer to 1 above)?
5) How is it placed in the preliminary waste route?
6) What aspects of the i-graphite handling can or should best be done manually, semi-remotely, or fully remotely.
7) How is the tooling system and its supporting systems influenced by the access route into the working area?
8) How is the tooling system impacted if the work is done in-air, under-water or in some other environment (e.g. inert gas)?
9) What other adjacent/related materials are there, how are they to be retrieved, and how might they influence the retrieval of the i-graphite itself?
10) What is the end-point of the project, specifically hand-over of material to the waste routes proposed to be used, and what issues are there arising from this waste routing regarding segregation of wastes and the working environment (see sections 7.2.1, 7.2.2 and 7.2.3)?

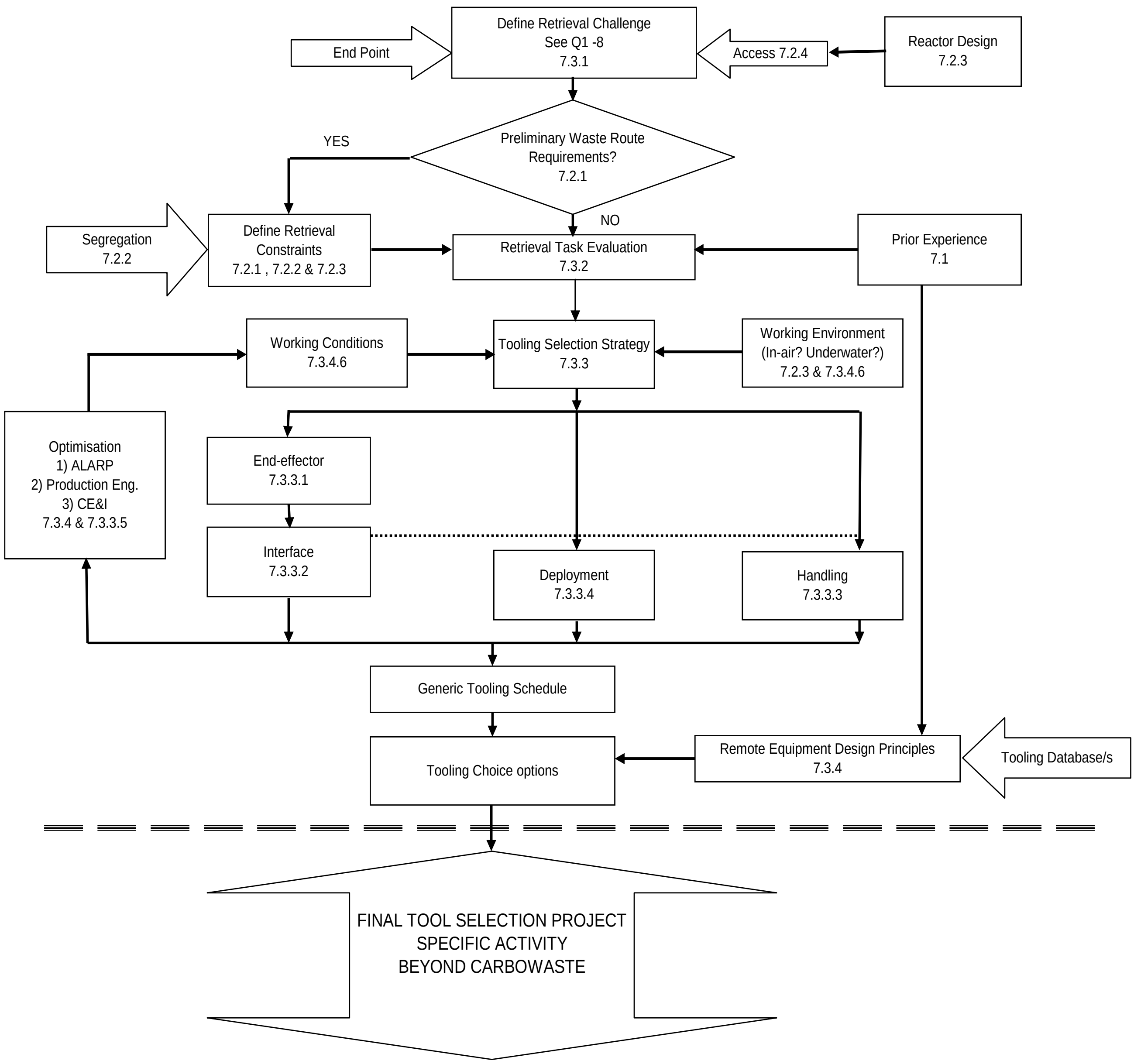

Figure 13 Flowchart for selecting retrieval tooling

It is therefore necessary to understand the design of the reactor (or other plant being decommissioned) and its operating history. The assessment of the condition of the graphite following its operating life in the plant will require an expert in this very specialist field and may also require significant efforts in measurement and sampling of the plant in its current state (as opposed to as-built). Exposure to elevated temperatures and heavy radiation fluxes over long time periods, particularly in CO_2 cooled reactors, has a significant material effect on graphite, and its mechanical properties can vary substantially as a result. This operational history also induces environmental conditions in the region where the graphite is located due to the activation and contamination of other structures and materials.

Whilst some i-graphite can be retrieved by manual intervention (“hands-on”) only, particularly in low-power experimental reactors, it is more likely in commercial power station reactors that some if not all of the retrieval activities will need to be carried out remotely. The following discussion is viewed from the need for remote operations. It should be recognised that some or all of the activities may be carried out hands-on in certain instances and consideration of opportunities to avoid complex remote operations should be encouraged.

Where i-graphite must be retrieved and segregated remotely, it is important to engage engineering capability with expertise and demonstrable experience in nuclear remote operations. At the time of writing there is no formal qualification available in this very specialist area, nor is there any universally recognised selection process to identify in potential candidates the attributes that will ensure their usefulness in this area of work.

In designing any remotely operated system, particularly for nuclear applications, it is a common mistake to start with the manipulator or remotely operated vehicle (ROV) which may look to be the most complicated and/or expensive part of the system. Rather it is best to resist this temptation and instead start the design process by identifying / developing the necessary end-effecter/s (the tools which will carry out the actual task) and then work outwards from there. Only after selecting end-effecters that will do the necessary tasks satisfactorily is it then sensible to design the systems that will support them. Where graphite removal is one of a number of sequential operations to dismantle the entire reactor, this philosophy should be applied to all retrieval activities, whether i-graphite or not and then a decision can be made on the following options: -

1) Attempt to optimise a common manipulation and access systems for all retrieval activities.
2) Optimise the manipulation and access system specifically for the i-graphite retrieval activity. This may require some earlier manipulation and access equipment to be discarded.

The final choice will need to take account of the more general context discussed in this report, especially section 6, as well as the detailed tooling design task.

7.3.2 Retrieval Task Evaluation

There are a number of factors to consider in evaluating the task of retrieving the i-graphite, in order to best inform the design process. The main factors are: -

1) Dose and contamination levels through the plant, and the corresponding preference or requirement to carry out operations remotely.
2) Dose levels through the plant (where they are too high for manual intervention) with respect to their influence on equipment selection and design.
3) Contamination levels through the plant, and their influence on equipment selection and design.
4) Working environment (air / inert gas / water) and its influence on equipment selection and design.
5) Programme – how the time required to retrieve the i-graphite and associated wastes interacts with other programmes, and how this influences retrieval strategy (with respect to ALARA/ALARP and handling rate).
6) Availability of services required, and the means by which they can be supplied at the necessary locations for the project.
7) Space required for plant installation, lay-down, support services, inactive and active maintenance, etc.
8) Suitable location for mock-ups of key areas of plant required for development and training of operators for the project.
9) Interfaces with waste routes for the i-graphite and any other wastes arising.

7.3.3 Tooling Selection Strategy

Whether or not the retrieval of the i-graphite is to be carried out remotely, the same basic system layout is likely to apply. A generic system layout of a remote retrievals system is shown in figure 14; note that most elements of the system shown may also broadly represent human activities using manually operated equipment, depending on the extent to which remote working is found to be necessary or preferable.

The following notes provide some more detail on the elements that typically make up a remote handling system; these should not be taken as exhaustive, but merely as a general guide, perhaps most useful for project managers in helping them to understand in general terms the issues that will need to be addressed by the remote operations specialist engineers whom they will need to employ.

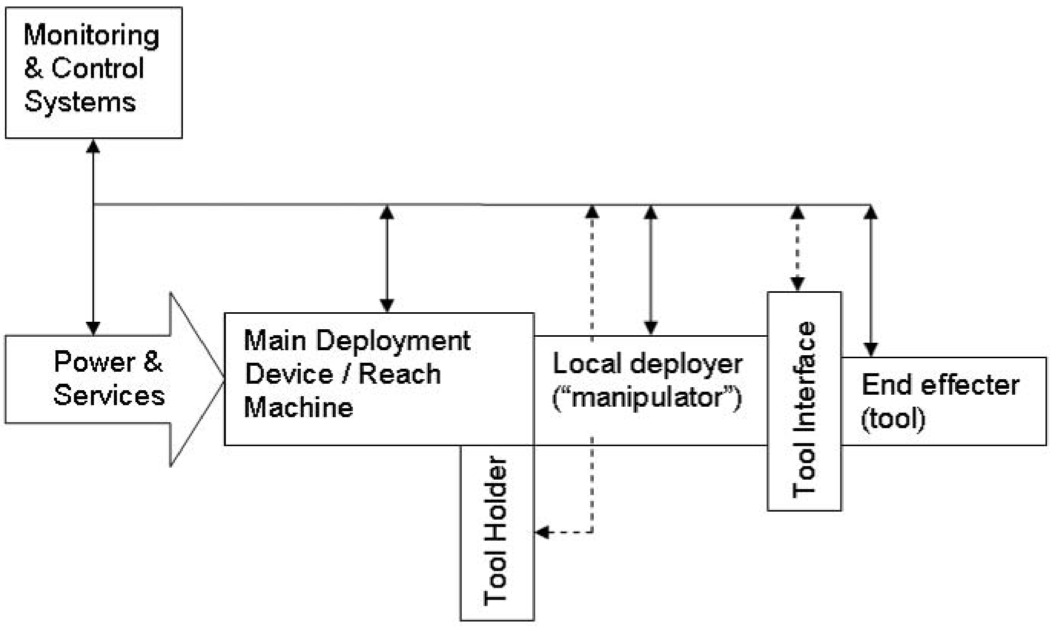

Figure 14 Basic Elements of a Retrieval System

The notes in the sub-sections below refer to the basic elements of the system as outlined above, and as such are intended to inform their selection, for both proprietary and bespoke equipment. As important as the functionality of each element are the interfaces an element may have with the other elements; failure to successfully integrate these elements into a coherent working system is likely to be just as debilitating to the project as a sub-optimal selection of elements.

7.3.3.1 End-effecters

Some examples of end effecters in the context of nuclear graphite retrieval projects are:

1) Inspection devices – CCTV and/or other transducers – to establish as precisely as possible the current situation within the area (e.g. reactor core, waste vault, etc.).
2) Sampling equipment to retrieve small specimens, again to assist in establishing the current status of the plant.
3) A grab of some kind, designed to grasp and hold the waste i-graphite components securely for transfer.
4) Some type of cutting equipment to clear away obstructions such as thermocouples, steel bracing, etc.
5) Some type of device for cutting or prizing open / cracking the graphite blocks themselves, as may be necessary where the blocks have been assembled in an interlocking matrix.

6) It may or may not be necessary to include in the list of end-effecters a “standard” jaw for general handling purposes.

Some dedicated nuclear remote handling tool databases exist, including one managemed by Doosan Babcock and discussed at the WP2 workshop by Lionel MacIver. However such data bases become quickly outdated and it is equally valid to refer to 'normal' tool suppliers' catalogues, and then adapt the tools for remote operation that satisfy the functional requirements specific to the application. Again, some past experience is invaluable in adapting proprietary tools for remote deployment.

7.3.3.2 Interface (mechanical support and power / control interface)

It may of course be that the chosen means of manipulating the end-effecters, which is next in the design chain, may have a standard jaw of its own, and this may be present in one of two forms: -

1) Mounted on a remote exchange fixture of some type, to allow it to be engaged and discarded remotely in the working environment.
2) A fixed and permanent part of the machine (manipulator or whatever) which can only be removed by hands-on maintenance; if this is the case, other end-effecters may be fitted with a “jaw friendly”, a bracket shaped to optimise the grip of the jaws on the tool in the required orientation.

In the case 1, this remote exchange plate can then be used for mounting other end-effecters; the manufacturer should be able to provide details of the mounting to allow these interfaces to be designed, and in at least some cases the manufacturer will be able to supply blank tool plates which can then be adapted to the user’s chosen end-effecters.

In the case 2, the jaw friendly needs careful design to ensure that whilst it provides firm location of the tool once the jaws have closed on it, it is nevertheless easy to engage and disengage the jaws from it when they are opened. Depending on the design of the jaw actuation, it may also be necessary to consider an emergency release of some kind.

Both these approaches have their benefits and disadvantages, as follows.

A remote tool exchange system will necessarily be more complex, and any additional complexity will always provide additional modes of failure and a greater challenge for maintenance / repair (and note that maintenance and repair are a more demanding area for nuclear remote operations equipment because of the likelihood of it being contaminated). On the other hand, the opportunity to pass services through the plate (such as power for the tool and signals to monitor its performance) means that trailing cables and hoses can be avoided, reducing one area of potential failures. Remote tool exchange systems have historically been more troublesome than “dumb” interfaces, using a jaw friendly, and some projects have had major issues with being unable to detach tools remotely from the manipulator arm; this is no doubt due to the necessarily more complex design.

A simple system of jaw friendly brackets, as described above, offers the main advantage of simplicity – the deployment device is rendered less complex, and therefore more reliable and easier to maintain, and the tools are cheaper to deploy and generally easier to exchange. The main drawback of the jaw friendly approach is that the tools power supply cables/hoses, and any signal cables, will trail from the tool and will thus be more exposed to possible damage.

Neither approach is universally better or worse, but rather each is more or less suitable for an application depending on the many variables that will differ from one project to another. Part of the answer to which is most suitable will come from what happens to whatever services the tools require, and the means by which these services are passed away from the worksite to their terminations. Tool exchange is particularly important in i-graphite retrieval noting the variety of morphologies in Table 10 in section 4.3.6.

7.3.3.3 Local Deployment Device – Manipulator

Many will understand the word "manipulator" as identifying a kinematically redundant (6 or more Degrees of Freedom – DoF) arm of up to a few metres reach, designed to carry out the near-range direct deployment of the tool(s). This general type of manipulator arm (with many variants in terms of power, mechanism type, payload, reach and dexterity) is common in the nuclear industry, but "manipulator" may also describe a more simple device, with perhaps only two or three DoF, and in some applications this is all that is required to deploy the tool adequately onto the task.

Although not always necessary, a 6 DoF manipulator arm is often required to present the tool to the work-piece with sufficient dexterity to allow the tool to successfully fulfil its function. This is generally not necessary where the tool function is very simple and so does not require any special alignment and/or motion; however the demands on any manipulator system can be reduced if "intelligent" tools can be devised which incorporate their own alignment and traverse systems so that deployment is simplified to the level of getting the tool to the worksite.

Manipulators can vary widely in power, size and sophistication, and are a separate subject in themselves; choice of the right manipulator for the task is a complex process, but for the purpose of this discussion it is most important to stress these two points:

1) The decision on whether a manipulator is required at all, and if so the specification it must meet, can only be taken once the end effecters / tools have been defined.
2) If a manipulator is required, then if at all possible an existing design (genuinely proprietary if possible) is far preferable to developing a task-specific machine.

Consideration can be given to using two or more simpler manipulators working in tandem rather than one complex manipulator which has to multi-task. Some cases may arise, perhaps in low energy reactors or commercial reactors 50 -75 years after shut down where one manipulator is manual and the other mechanical. Both set up and retrieval operations need to be considered in making the choice with consideration being given to ALARA.

7.3.3.4 Deployment Device – Reach Machine

A number of options are available to transport the tooling and its local deployment device (manipulator or other) into the working area, some as proprietary products and others as tried and tested design approaches. These are:

1) Remote dismantling machines mounted on RPV (e.g. WAGR), typically – but not necessarily – on a rail system.
2) Hoist systems:
   a) Mounted outside RPV or bioshield (e.g. GLEEP)
   b) Mounted and supported inside the RPV
3) Rigid extending mast (e.g. Vandellos Silo emptying)
4) ROV(s), possibly in combination with one of the other options
5) Manual:
   a) Direct hands-on working (low dose situations)
   b) Indirect using set up of jigs and retire to a distance (i.e. semi-remote). This may be the optimum approach if doses allow and if, for example, the number of blocks to remove is so large or they are so heavy that pure manual working is not efficient or safe (e.g. GLEEP).

7.3.3.5 Interface – Services Supply

Most end-effecters will require services of some kind, usually power but often signals too, passed back and forth to monitor and control the functioning of the end-effecter.

Where a remote exchange system is used, some provision for such services is generally included in the tool plate, which services (hoses and cables) are then passed down through the arm, providing mechanical and chemical protection to them from the working environment as noted above.

Where a simple jaw friendly is used, the associated cables and hoses will inevitably be free-trailing, which leaves them more susceptible to damage but also more exposed for ease of maintenance/repair.

The overall size of the reactor space needs to be considered bearing in mind the increased depth of operation as the graphite core is reduced in height. Thus there will be a differentiation between devices that sit on the top of the reactor and devices, such as ROVs or lowered platforms that work on or close to the surface of the i-graphite.

Once away from the immediate working area, the services must pass back to wherever they are to be terminated for supply of power and exchange of monitoring and control signals.

In the case of a remote exchange system where service couplings are included, the services will then pass down inside the deployment device (manipulator) and exit at the base, to then be routed along the main deployment system along with whatever other services there are.

In the case of a jaw friendly, this will generally be achieved by cabling to the location of the tool holder from which the deployment device (manipulator) selects its tools, and this may be part of a greater deployment device (a reach machine of some kind to access the plant), or it may be installed in the plant where this is possible and preferable. The connectors used on the end of the flying lead to connect to the tool holder should be designed to be coupled and de-coupled remotely, using the manipulator, so that in the event of failure either of the tool or its flying lead, a replacement can be posted in and installed on the tool holder remotely, using the manipulator.

7.3.3.6 Tool Holder

Even when only one tool is to be used, it makes sense to have a tool holder of some sort, so that if the tool fails a replacement tool can be picked up remotely and work continued. In most cases, several tools are required and the tool holder will have several stations on it to hold the different tools. In situations where a large number of tools are to be used, it may be necessary to have some form of powered exchange system to present to the manipulator the tools needed at the time out of the wider range available; such systems are often in the form of a powered carousel.

As noted above, where a simple jaw friendly system is used for the manipulator to hold the tool, any services to the tools will need to be via the tool holder; if a powered tool holder is required for presenting large numbers of tools to the manipulator, providing services in this way will become more complicated, and so will strengthen the case for a remote tool change system that incorporates services passed up through the manipulator arm.

7.3.3.7 Monitoring and Control Systems

The requirements of the monitoring and control systems will be largely defined by the selection of the tooling and deployment systems described above, but they are no less important to design competently for optimal working.

As much data as possible should be captured at the work-site and relayed back to the control area; not all data need be used (and failure of some data acquisition systems might later be accepted without stopping work), but data that is not captured can often be impossible to estimate with any useful degree of accuracy. The most obvious source of information to relay to an operator of a remote system is imaging, either by line-of-site or by the use of CCTV images; for totally remote working, stereoscopic CCTV ("3D") provides major improvements in productivity and reductions in operator error, but the system must be competently designed and set up to have the desired affect and avoid adverse affects. Other sources of useful information can be strain measurements, current consumptions etc. taken at key positions to indicate the forces the machine is experiencing; sound systems with microphones at the worksite and speakers at the operating area are also surprisingly useful in enhancing an operator's perception of a remote working environment.

Careful consideration is needed to design the man-machine interface, using ergonomics and human factors expertise to maximise the operators' effectiveness. This includes location and type-selection of display and input devices to maximise operator effectiveness and minimise fatigue. It also includes decisions on what data to present to which members of the team such that the main operator has enough information to work effectively but is not swamped, and his/her assistant/s are able to monitor other data and advise/assist in the control of the system as a whole. All these considerations should be made in conjunction with design of the operations team structure to optimise the operations system as a whole.

Careful consideration is particularly important when deciding control interfaces for the tools and, if present, manipulator; some interfaces may be superior in the level of control they offer but are more arduous in use, and may be better augmented or even replaced by less sensitive but also less arduous interfaces. Similarly, the level of "immersion of the operator" should be carefully balanced – a high level of immersion allows very efficient working, but is arduous and efficiency will fall off more quickly due to operator fatigue.

7.3.3.8 Decontamination and Maintenance Facilities

Although not shown in the generic system diagram, Figure 14, it is vital that appropriate facilities are provided to decontaminate and maintain the remote equipment so that it can be kept in operation. When the retrieval project is complete, it is obviously better if the equipment used can be retrieved and decontaminated either to be used again on another reactor or on another site or, if possible, to be cleaned to free-issue standards and scrapped (rather than becoming LLW or ILW).

7.3.4 General Design Principles for Remote Systems

The following is provided as a basic guide to designing remote handling systems for i-graphite retrieval in the context of in-reactor operations as is the case for i-graphite cores. As noted previously, this is a specialist area in which suitably experienced staff are particularly important to the success of a project.

7.3.4.1 Design for Reliability

Access to the remote equipment may be restricted once it has entered active service. This will not just be due to the general need to minimise down-time, but also due to difficulties in retrieving the equipment and working on it. It is therefore even more important than usual to design the equipment in such as way as to minimise the potential for failures, and the following are key elements in this: -

1) Wherever possible use proprietary products as the building blocks for the system.
2) Where bespoke devices must be procured, wherever possible these should be based on similar equipment used successfully on previous occasions, and designed using well-proven and readily available components.
3) Minimise complexity – each element of the system, and the system as a whole must be as simple as is possible whilst still being capable of achieving the functional goals. Consider the function of the i-graphite removal task; dismantling not high-precision engineering; with variable morphology from dust, through fragile to massive stable blocks.
4) Design equipment to be robust, with generous safety factors applied to loadings, particularly where these are less certain, to minimise the possibility of structural failure.
5) Train operators selectively and thoroughly, and if possible rehearse the retrieval operations in a mock-up facility to minimise the likelihood of equipment failure due to operator error or misuse of the equipment.

7.3.4.2 Design for Retrieval

If equipment is to be operated remotely because of the prevailing doses, it follows that retrieval of the equipment in the event of failure is likely to have to be carried out either partially or completely remotely. The equipment must therefore be designed with this additional challenge in mind, and following are some of the tried and tested features used to facilitate remote retrieval of various types of equipment: -

1) The transmission of wheeled machines, whether on rails (crane-like or bogie-like machines) or free-ranging, should be designed such that on loss of power clutches are disengaged and brakes are off, so that the machine is free-running; this allows it to be more easily dragged back to the decontamination and maintenance area. This is also applicable to any turntable devices unless their free-running would present other problems.

2) Umbilical cables should include strainer wires sufficient to bear the load of dragging back the machine if it becomes disabled; for large machines that cannot be drawn back manually, the cable reeling machine should be designed to be able to wind in the umbilical for retrieval of the machine.
3) Where possible (such as on gantry machines), redundant drives should be provided, with suitable transmissions to allow them to take over for recovery in the event of failure of the main drives; these back-up drives should be geared for very slow operation, to discourage operators from using them to continue work and thus risk failure of the back-up drive and so total loss of mobility.
4) Manipulator jaws and any other part of the machine that grips or holds waste material (of potential high activity including activated steel components mixed with the i-graphite) should be designed such that the waste material is released on power failure, and so is not drawn back with the disabled machine (which could cause elevated dose rates in the decontamination and maintenance area), unless this would present other problems.
5) Where major movements of a deployment machine are achieved using hydraulic cylinders, the controlling valve/s should be in a man-accessible area so that failure of the control system does not prevent emergency retrieval (which can then be carried out using hand-operated pumps if necessary).
6) Where the retrieval aperture is so small as to require this, manipulator arms can be designed to fail in such a mode that their joints are slack/compliant, and so will deflect easily allowing the manipulator to conform to the aperture through which it is drawn (unless this would present other problems).

The above issues need to be considered whatever, is retrieved from the reactor whether i-graphite. Specific features to i-graphite are commented upon throughout section 7.3.

7.3.4.3 Design for Decontaminability

As noted above, access to the remote equipment for maintenance may be constrained, and a contributory factor to this is levels of contamination on the equipment or other entrained materials and the ease with which they can be reduced. The equipment must therefore be designed to be as easy as possible to decontaminate; some important points in this are: -

1) Early evaluation of likely contaminants and resulting selection of appropriate decontamination technique/s (swabbing, hosing down, spraying, etc.) and chemical agents to be used (whilst considering of the effluent disposal routes proposed).
2) Selection of exterior materials compatible with selected decontamination techniques and agents.
3) Sealing standards (generally IP54 to EN 60529 as a minimum) to be applied to the design appropriate for the selected decontamination technique/s and agents.
4) Design of equipment shape to minimise or eliminate contamination traps such as concave corners or voids where dust and other loose materials may settle and become difficult to dislodge.

In the case of i-graphite, experience at GLEEP was that the health physics monitoring equipment used (hand-held on that project because radiation levels were low) needed to protected against the ingress of graphite dust (greasy) that is found on the surface of the graphite blocks).

7.3.4.4 Design for ease of Maintenance and Repair

As noted above, access to the remote equipment for maintenance is likely to be severely constrained and residual doses arising from contamination remaining on the equipment may limit the time operators can spend working on it. It some cases it may also be necessary for maintenance operatives to wear personal protective equipment (PPE), perhaps even a full PVC suit and respirator, and this will further contribute to the difficulties in maintaining and repairing the equipment.

It is therefore most important that maintenance and repair are made as easy as possible; a key aspect of this is modular design, so that individual sub-assemblies can be readily removed for further decontamination (if required) and repair or maintenance. Spare modules may also be procured to exchange on the machine to minimise down-time (the failed module then being repaired off-line).

It is also important to design carefully the fastening systems for any modules or individual parts that may need to be removed during the life of the machine. Hex-headed bolts are preferred to hex-socket (“Allen”) heads so as to avoid contamination traps, but they can also be cone-headed to make for more rapid engagement of sockets or ring-spanners during removal. Fasteners requiring more dexterity and/or special tools are to be avoided.

In particular, fasteners such as circlips and split (“cotter”) pins should be avoided as they can present sharp edges, and indeed maximum effort should be made to eliminate all sharp points, corners or edges as they can be particularly hazardous to maintenance operatives in a contaminated environment.

Areas for maintenance need to be allocated, preferably close to the operational area, to facilitate maintenance operations associated with the above activities. The design of the reactor building envelope, the charge face and other adjacent areas already vacated of redundant equipment need to be reviewed for potential locations.

7.3.4.5 Design for Remote Operability

Designing equipment to be operated remotely requires a different approach when compared with conventional equipment. Due consideration must be given to the level and quality of information fed back to the operator to allow him to understand how the equipment is operating, and the extent to which he/she is able to control that operation; both these will be somewhat limited compared to the operation of normal equipment where the operator is adjacent to it.

As a simple example, consider the use of a conventional overhead crane compared with the operation of a remote deployment machine: with the former the operator is generally almost directly below the crane and has a clear view of the entire machine and its surroundings, and so is able to operate the machine safely with travel speeds of perhaps 50 m/min; in contrast, the operator of a remote deployment machine may be able to see very little even of the machine itself, and its main travelling speed is likely to be limited to around 4 or 5 m/min to allow the operator time to continuously assess its juxtaposition with other plant in real-time, and so avoid collisions.

On a more detailed level, where pieces of equipment are to interface, e.g. one piece lands on and combines with another, suitable lead-ins need to be designed into the equipment and remote coupling systems included, to account for the fact that, unlike the situation with conventional equipment, there will be no operators at the worksite to guide the equipment together and make up the couplings by hand. Failure to properly design such interfaces adequately is a common cause of extended operation time and damage to equipment when operating remotely.

The paragraphs above consider operations in air; reference has been made in section 7.3.4.6 below for design for remote environments such as retrieval from underwater. The issue of water shield allowing proximity and good line of site visibility directly above the workface was cited by EdF at the Work Package 2 workshop at Gateshead as an advantage.

Some reactor designs, and Windscale AGR was an example, do not have unrestricted vertical access to the graphite. In the case of Windscale AGR this was because reactor pressure vessel contained an internal corbel, which meant that some i-graphite could not be accessed directly from above but required some lateral movement to reach areas under the shadow of the corbel.

7.3.4.6 Design for the Remote Environment

Remotely operated equipment is designed to access areas where humans cannot (or should not, for ALARA purposes) go; it follows that the environment may have some significant differences to man-accessible areas, and may also present some related challenges to the remote retrieval equipment.

In i-graphite retrieval projects the most obvious challenge is the radioactivity present; this can be disruptive and damaging to electronics in medium dose levels, and in more excessive doses can also damage materials, particularly non-metallic materials such as plastics used for electrical insulation, hydraulic hoses etc. Accurate and realistic assessment of doses is required, along with assessment of proposed equipment by suitably knowledgeable and experienced experts to assess the possibility of radiation damage.

Secondary wastes arising from the equipment must also be considered. Any part of the equipment that cannot be decontaminated to free-issue levels will become part of the LLW or ILW inventory. The design of the equipment must therefore take account of this not only in maximising decontaminability (see 7.3.4.3) but also in compatibility of the equipment to the waste routes to be used.

Compatibility with active effluent treatment and disposal routes may limit or completely preclude the use of hydraulic fluids, accepting that a leak-free operating life is highly unlikely, if not impossible. Early agreement on effluent disposal arrangements is therefore essential to inform design decisions on the remote equipment.

7.3.4.7 Working under-water

Finally, another challenge the remote environment may present is immersion in water; whilst most projects involving the retrieval of i-graphite to date have been performed in air, some have been or are planned to be carried out underwater to provide shielding that then allows a closer approach by the operators. This report has discussed in some detail the proposal to retrieve graphite under water at Bugey in France and compared this approach to that in air (particularly sections 6.2.4 and 7.2.3).

The presence of water will affect the design of the equipment, not only in requiring it to be submersible but also to be resistant to the corrosive affects of water, depending on the specific water chemistry expected on the project. In the case of the submersion it will be necessary to specify the depth and work in accordance with the code for specification for degrees of protection provided by enclosures IP code EN 60529:1991, for example, for which condition IP68 is anticipated for situations in which i-graphite is being retrieved from a commercial reactor.

Being submerged in water may also present additional challenges to retrieval of failed equipment compared to being in air, although this may depend on the level of man-access that would otherwise have been possible with the prevailing dose rates. An advantage of using water is the impact of the shielding affect, which tends to enable the use of simple manually operated tools, thus significantly limiting the potential for failed equipment. Depending on the general radiation levels in the reactor pressure vessel it may be possible to lower the water level and the lower the platform upon which the operatives work as the core is dismantled and lowered in height. If this were not possible the issue of handling long reach tools might need to be considered.

7.4 Modelling Strategy

As an input to the Work Package 2 workshop the following types of modelling were identified as of potential benefit to CARBOWASTE; specifically with respect to supporting retrieval and segregation needs. These would provide useful tools for the CARBOWASTE toolbox.

1) Nuclear physics modelling - Modelling of the in core behaviour based on inventory and irradiation history to predict post irradiation inventory and its distribution.
2) Chemical/molecular modelling - Modelling the structure of the graphite, giving consideration to factors such as chemical oxidation and graphite voidage and Wigner energy
3) Structural Modelling - Modelling of the stresses in reactor and used to consider structural issues, for example thermal and oxidation impact on structure, in core stresses, locking of bricks, understanding importance of condition of bricks to retrieval
4) Mass Balance/Flow Sheet - Modelling of overall mass balance and inventory of the reactor system; particularly to consider the origin of C-14.
5) Operational Research / production engineering – Modelling of the processes such as retrieval operations to consider potential throughput of unit operations.
6) Visualisation – This is a generic group of models, but could be used for through core mapping, 3D/Virtual Reality (mock up) and could allow the results of the other modelling techniques to be presented and understood
7) Environmental – Modelling of discharge dispersion or dust generation
8) Heat Transfer / CFD - Evaluate the performance of specific unit operations or systems, in terms of heat transfer and fluid flow.

The workshop confirmed that these areas were all relevant to CARBOWASTE. However, to focus attention on areas of priority for retrieval and segregation and where further development was considered necessary, the following specific areas were recommended for further study: -

1) mass balance modelling (including nuclear physics for inventory),
2) structural modelling,
3) operational research modelling and
4) visualisation.

It is recognised that a considerable amount of modelling work exists and the group felt that it was important that the first activity under task 2.3 should review current capability. Task 2.3 has therefore proceeded as follows: -

1) develop an overview document that outlines the key types of models, their relevance to the CARBOWASTE project and where possible examples of where they have been applied.
2) testing or development of the most relevant tools.
3) issue modelling and diagnostics report
4) update this report based on the modelling development output.

It was considered that it would be extremely unlikely that desk top modelling would eliminate the need for physical modelling, which is very important for safety training purposes as well as supporting and demonstrating designs developed initially with desktop modelling. Examples of physical models of AGR reactors located at Gateshead were viewed by the workshop attendees. The strategy considers that desk top and physical modelling should be considered as complementary activities.

8.0 Integration Issues

This section is a summary of the integration issues that have emerged as a consequence of this Work Package 2 study. They are brought in to prominence at this point to remind the reader that the decommissioning of a graphite moderated nuclear commercial reactor in particular is a complex integrated project involving many stages and facets. The key areas are the policy and regulatory approach, the reactor dismantling itself and the waste management activity. These three processes are themselves integrated.

8.1 Planning, Safety Case and Implementation

The progression of a project from the plant operator deciding to do it to finally achieving an agreed endpoint when perhaps there is some residual or perhaps no radiological harm (a requirement for de-licensing of a nuclear site in the UK) has three recognisable stages. Linking retrieval and segregation in to these was summarised in Table 19.

When the decommissioning plan is produced for a nuclear power station, what to do with the i-graphite should be an important component of this plan. Options may be left open because of lack of waste disposal routes for graphite, a gap which CARBOWASTE is endeavouring to close. It is hoped that the information in this report can be used to feed the expected i-graphite work breakdown structure, criteria and potential solutions to retrieval and segregation in to the plan, which will be produced before detail consideration of the specifics of i-graphite retrieval have been considered.

The plan will lead to at least bounding conditions and possibly some options for i-graphite retrieval and segregation and tasks will be required in the work breakdown structure that lead to one or more safety cases being produced, which themselves bound the implementation opportunities. This report has endeavoured to provide a broad range of check lists and analysis that will assist the safety engineers to define the key features of the challenge. It will be enhanced when the modelling activity currently being undertaken as Task 2.3 of Work Package 2 by NNL is complete. This report will be updated to comment on the additional modelling tools available.

For implementation it will be necessary to get in to detail about the plant, the graphite condition and the preliminary waste route that is beyond the scope of this report. Never the less in combination with Work Package 1 output this report has endeavoured to cover the range of issues that the implementation engineer will need to consider and again the checklists are commended. Section 7.3 of this report gives guidance on a strategy for selecting tooling and lessons learned from previous projects have also been referenced, which it is hoped will be a useful staring point. But the need to consult the supply chain, whose ability to innovate and provide experience and solutions should never be overlooked.

The final comment is that in terms of integrating the planning, safety case, implementation sequence all participants should be involved at the earliest opportunity in the process including operators, regulators, supply chain and various stakeholders. This report, especially in section 5.0 and figure 3 has strongly advocated taking a staged solution engineering approach to the development of an i-graphite retrieval and segregation solution.

8.2 Reactor dismantling process

The global reactor dismantling process is not in the scope of Work Package 2 yet it has inevitably been referenced many times through out this report. Both in reference to he Windscale AGR project and the Bugey project being undertaken by EdF, the fact that i-graphite is not necessarily the whole of the story has been stressed. It will therefore not be surprising that many of the principles suggested in this report as being appropriate for i-graphite retrieval and segregation are equally applicable to the retrieval of other materials from the reactor. The corollary is that tooling that may have been developed for some other purpose may be used or adapted for use on graphite. The study has identified four key issues for tooling:-

1) A carrier device which could be:
    a. A suspended platform (perhaps crane deployed) e.g. Bugey
    b. A remote dismantling machine mounted on the RPV e.g. Windscale AGR
    c. A remote operated vehicle (possible option for Magnox or retrieval of unconditioned waste (Vandellos)
    d. A submarine (has been used for underwater inspection of BWR but not for i-graphite)
2) The arm or tool manipulator
3) The tools, associated processes and interfaces
4) The control system

With the possible exception of some specific tools suitable for i-graphite core blocks like the ball race mandrel or drill and tap, most of these elements are of generic use and can be applied to any reactor dismantling activity.

So i-graphite tooling development should continue in parallel and association with general reactor tooling development. The reactor decommissioning engineer will not be specifically concerned about retrieving i-graphite; his challenge is how do I empty the reactor before I dismantle the reactor pressure vessel.

8.3 Integrated Waste Management process

The Integrated Waste Management process is the key feature of the CARBOWASTE project. Instead of considering retrieval and segregation in isolation, the approach has been to consider the whole i-graphite management process form retrieval to disposal.

This report has been written before the CARBOWASTE Work Packages 3, 4, 5 and 6 are complete. The Work Package 2 team has therefore liaised very closely with the Work Package 1 team and inputted in to a higher level review of retrieval and segregation, which related to two key aspects

1) Understanding the reactor designs and back ground experience of graphite
2) Developing and understanding criteria for assessing retrieval and segregation options.

The first of these have been reported in the Work Package 1 Review Reports [7] for each country and should be referred to in addition to this report. The second area will be reported by Work Package 1 under its output to Task 1.6 Retrieval Criteria [14]. This latter task has led to the development of the concept of the Preliminary Waste Route that forms a pivot point between the requirements of the integrated waste route and the challenge for the retrieval and segregation engineer. This report has therefore included a summary of the Work Package 1 deliverables in section 5.2. It should be noted that these will continue to be refined by Work Package 1 throughout the CARBOWASTE project and the reader should be aware that some information in this report may subsequently be superseded or expanded.

9.0 Discussion, Conclusions and Recommendations

9.1 Discussion

9.1.1 Value and update of knowledge

The value of this report is only as good as the information available and knowledge at the time that it is written. It has some considerable advantage in that retrieval and segregation of i-graphite is not an unknown activity but has been demonstrated on other projects on research / test / demonstration reactors although generally not on the scale of most commercial reactors. It is desirable that future lessons learned can be added to the state of knowledge and this could be achieved at two levels. Firstly by international agreement, either within Europe or beyond, a means of continually updating the CARBOWASTE toolbox would be beneficial. Secondly, within operating organisations, especially those who have a fleet or reactors such as EdF and NDA, this report could be adapted to add on in-house information that is not more generally available and a structured approach to recording i-graphite retrieval and segregation information and relating this to knowledge transfer, education and training. In the UK for example there is a Decommissioning Waste Forum sponsored by the Environment Agency that encourages the collection and collation of waste management and decommissioning data and it has been proposed that the UK users incorporate a i-graphite sub-group within this vehicle.

9.1.2 Content of the report

The prime objective of the CARBOWASTE project has been to produce a toolbox of techniques that will aid users in the future to develop best practice i-graphite management solutions. By calling upon a broad range of experienced persons from operators, contractors, academia, laboratories research with a view to engineering application has been possible.

In the case of retrieval and segregation it was recognised that this was a prerequisite to the waste management steps that might follow and that the retrieval step itself has not been specifically covered in documents on i-graphite management such as the IAEA TecDocs.

So this report has been seen as a tool box compartment specific to retrieval and segregation.

The report is largely based on the output of a two day Work Package 2 workshop held at Gateshead in July 2009 involving Work Package 2 beneficiaries some from other work packages and students. But to arrive at this workshop a staged approach was adopted. Firstly, some members of the Work Package 2 team engaged with Work Package 1 to develop upper level criteria and examine outline retrieval options against these criteria as part of Task 1.6. Also many Work Package 2 beneficiaries were engaged in writing reports for the Work

Package 1 Review Report that included information on key reactor designs, graphite and previous retrieval experience. A Data Pack was produced by the Work Package 2 legacy waste tasks leader that included graphite retrieval specific information and introduced strategic issues and other criteria that might be used in retrieval and segregation base on past experience. This Data Pack was reported on the SINTER Data Base for Work Package 2 beneficiary use and was a key input to the workshop. The workshop report was also published on the SINTER data base.

This report has therefore drawn upon all these sources. It is not intended to be a re run of these earlier reports but pulls together the key issues that will be considered in developing a retrieval segregation solution that both enables the i-graphite to be removed from the reactor, the prime objective, but also defines the relationship with the integrated waste management strategy.

With the view of it also acting as a tool box compartment for retrieval and segregation, the report has been presented as far as possible with a considerable number of tables, some of the figures and a number of sections that can be considered as checklist that the engineer developing his process can call upon to ensure that his design is complete. In addition a work breakdown structure at an upper level has been produced and linked to the checklists.

The goal of this tool box compartment is to help build up i-graphite retrieval projects in a structured way so that those with specific problems can design and evaluate specific solutions knowing that they have reached their solution by way of a logical process, that they have taken note of existing experience, that they understand the position of their project in the overall scheme of things, not only graphite management but in terms of the global decommissioning context. The tool box compartment cannot make specific recommendations as each solution will be specific to their reactor design, their operator, their regulator and may be even their country of origin's policies.

The document will also be a useful guide to students or new entrants to the industry who may be interested in i-graphite retrieval. It is particularly hoped that researchers who are developing new technologies will find it valuable in terms of focusing on the key issues as the report contains a high element of industrial experience and practice.

There is lesson learned by many industrial companies who develop new products and processes. It is often the case that the research and development focuses on the novelty and innovation and goes to great lengths in the laboratory to get the new process to work only to find that it is the existing practice that fails when the new process is applied in the field. That is why this report also states a lot of the obvious; it is the obvious that will go wrong, if ignored!

9.2 Conclusions

1) A generic retrieval and segregation process has been defined

2) Definition of the Preliminary Waste Route by examination of the Integrated Waste Management Strategy is fundamental to further development of the generic retrieval and segregation process

3) The parameters that input to the design of each operation in the generic retrieval and segregation process have been defined.

4) Consideration of prior experience and workshop analysis have led to a comprehensive review of the issues that affect both the project design and the engineering design. Over 30 checklists have been produced that allow a designer or reviewer to check the content and comprehensive coverage of the design activity.

5) These checklists have been related to a work breakdown structure for development of a retrieval and segregation design.

6) Criteria for selection of retrieval and segregation processes have been identified at both higher and lower level and applied in a demonstration MCDA.

7) A comparison of the processes of retrieval in air and underwater has been undertaken.

8) Feasible processes exist to retrieve i-graphite from reactors and unconditioned waste stores but the issue of removing jammed blocks from reactors in which graphite has expanded has not been finalised.

9) A good appreciation of the design and strategic issues that affect the main commercial reactor designs, Magnox, UNGG and RBMK has been achieved and engineers working on these different systems have benefitted and should benefit in the future from the cross fertilisation of ideas.

10) A process for selecting tooling has been developed

11) Several younger engineers relatively new to the industry have taken part in the project.

9.3 Recommendations

Report Finalisation Recommendations

All Work Package 2 beneficiaries should address the comments in this report to see if they can add to or clarify points made or check for errors in interpretation

The document should circulated to other Work Package leaders who might identify specific issues related to their projects.

It would be particularly useful for the table of properties prepared by Work Package 1 to be revisited

Other recommendations to be confirmed or added to

Specific Operators, States or EU should consider ways to maintain and update the knowledge base on retrieval and segregation of i-graphite. (This comment could equally apply to any aspect of i-graphite waste management). Ownership and funding for such on-going management needs to be identified.

Consideration is given to developing this report in to an IAEA TecDoc

The report should be reviewed by the modelling team to cross check the optimum output of models.

10.0 References

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11.0 List of Tables

12.0 List of Figures

13.0 Appendices

APPENDIX 1 – DEFINITIONS

Access: See the specific discussion of why access is required in section 3.2 (2).

Anti-lift device: In the context of CARBOWASTE and noted for Wylfa NPP, a component placed in a reactor to prevent graphite blocks being elevated due to upward forces caused by other operations in the reactor.

Bioshield: A name (usually used in UK) for the concerted shielding around a nuclear reactor or some components associated with it such as steam generators. May be also be referred to as the reactor shielding (e.g. in the Lithuanian reporting).

Bottleneck: an activity that creates a critical path, for example during retrieval, such that other in-line processes are unable to operate at their full capacity.

Buffer Storage: for the purposes of retrieval and segregation, buffer storage is specifically used to allow for local control of the retrieval and segregation process, for example i-graphite may be retrieved at a different rate to that which it can be segregated or sentenced and buffer storage may be required to allow for this miss-match. Buffer storage will be considered as a logistical activity that may occur than dispatch. Any storage after ***dispatch*** will be considered part of the ***waste disposal route***.

Conditions for Acceptance: The specified requirements of a waste disposal or storage facility to for the form of waste that it accepts. This may include details of radiochemical and chemical content, form of processing, packaging, transport etc.

Cores: Whilst the core includes the graphite moderator, for the purposes of this document it is also intended to include other graphite in the core region such as thermal shields, reflectors and thermal columns.

Dispatch: the point of handover of retrieved and segregated i-graphite or associated waste to the packaging/conditioning process.

End User: An organisation such as a Utility/Owner, Regulator or Waste Disposal Authority who will ultimately be responsible for the decommissioning and waste management strategy of a facility containing i-graphite.

End effecter: This is the device that actually interfaces with the i-graphite or associated materials and undertakes some function such as connection, modification or lifting and placing.

Feasibility, flexibility and value: implies that retrieval and segregation techniques proposed can be reasonably engineered using proven technologies that are safe in application, of economic benefit, socially acceptable and not detrimental to project risk. Whilst innovative or new ideas are to be encouraged they are unlikely to be accepted or applied unless they can substantiate compliance to regulators, fund managers and implementation contractors/risk owners.

Fuel Element Debris: a term used at Berkeley and Hunterston A to refer to various i-graphite components such as fuel element struts or sleeves and that may in some cases be mixed with other fuel element components and thermocouples for example. The radioactive source term of the waste from a radiation point of view may be dictated by the other activated components rather than the graphite.

Furniture: Term sometimes used for objects such as baskets and steel frames to aid placing of position of wastes in a waste box.

Gangue Materials
These are miscellaneous materials, not graphite, that are integrally associated with the graphite before or during its handling in the reactor. They could be particulate material attached to the graphite or in the pores or cracks, pieces of monitoring equipment such as thermocouples that become mixed with the graphite during handling, contamination carried around the reactor circuit or during decommissioning similarly mixed with or attached to the graphite, other materials in there reactor that may be removed at the same time, including insulation (e.g. asbestos). Gangue Materials may vary significantly from reactor to reactor, design to design and between commercial and test reactors.

Gated Process/Gate: A project management process where review hold points are introduced generally to manage the level of risk or to enable effective project re-direction as more information is available to the project. The Gate will have a number of acceptance criteria of qualitative and quantitative nature that will need to be achieved and has often an element of timing also.

In the case of retrieval and segregation the term “Gate” is applied to the Preliminary Waste Route because it is considered feasible to define this route in such a way that retrieval and segregation can proceed even though a number of alternative waste route options may or may not be available downstream at the time retrieval and segregation takes place. In defining the criteria for the Preliminary Waste Route it is therefore essential to enable retrieval and segregation to take place whilst at the same time avoiding foreclosure of reasonable downstream waste route options. In the case of GLEEP the Preliminary Waste Route allowed i-graphite to be retrieved from the reactor even though there remained choice of downstream waste route between further processing and dispatch to a Low Level Waste repository; one of which options could have been withdrawn.

Generic Process: Is generally applicable to any reactor design, nuclear site or State.

i-Graphite: irradiated Graphite

Immersion of the operator: This refers to the intensity of concentration of a remote working operator in the work that he is doing. If not carefully managed this can have impact on the health and performance of the operator both on the task itself and also afterwards when the operator leaves the working area.

Integrated Waste Management: The process being developed in Work Package 1 that considers the optimisation of all of the process steps from identification and retrieval of irradiated graphite through its treatment, re-use, recycling to its final disposal.

Interim Package: in the context of figure 1 and interim package might be a basket to lift retrieved or segregated graphite components out of the reactor. The interim package may be re-useable or could become furniture inside a waste disposal box. It is not expected that it will normally be a disposal container, however, in the interests of innovation any practicable option will be considered.

Jaw Friendly: A flexible device for interfacing a variety of tools in various orientations to the manipulator. The purpose is to maximise and facilitate the number of tools(end-effectors) and orientations that can be used and changed during operations. It is suggested in section 7.3.3.2 that this might be some form of bracket that fits in to a jaw on the manipulator. The use of this term does not rule out any other engineered device of similar purpose that achieves the desired effect.

Legacy Waste: Waste arising from nuclear reactors already shut down, well in to operational life or already established designs as opposed to waste produced by future designs of reactors such as those covered by Work Package 2 Tasks 2.4 and 2.5. Legacy waste is covered by Tasks 2.1 to 2.3.

Letter of Compliance: A practise in the UK wherby in the absence of a repository a waste disposal authority can accept a waste form on the basis that it will comply with the requirements of an envisaged generic repository design.

Manipulator: In the context of technology selection for CARBOWASTE this is any device that can handle and direct tools to a work face; it may include on the one hand a complex multi degree of freedom robotic device or on the other a simple winching system on rope. In a low dose system where human intervention is possible, the manipulator by the human.

Operational waste stores: The preliminary resting place for any graphite components, such as sleeves and fuel element debris already removed from a nuclear reactor as part of its operational process. This storage is temporary and precedes any further waste management.

Preliminary Waste Route: The immediate route for i-graphite and other materials removed from a reactor or other non-conditioned waste store but prior to its conditioning/packaging for transport away from the vicinity of the reactor to an interim store or disposal site.

Retrieval: in the context of CARBOWASTE WP2 retrieval for legacy waste is the processes necessary to gain access to and remove i-graphite from a nuclear reactor vessel or associated facility that stores un-conditioned i-graphite.

Segregation: in the context of CARBOWASTE WP2 segregation is the process necessary to separate any materials incidentally handled as a consequence of the i-graphite retrieval process at the time the i-graphite is retrieved that may be inappropriate to be ***dispatched*** together with the graphite. It does not include the segregation of contaminants in the graphite that are the subject of WP4 or WP5.

Reactor Dismantling Activity: comprises activities that may take place before or after i-graphite retrieval in a planned sequence to fully dismantle a nuclear reactor. Some of these activities may be necessary before the i-graphite can be accessed and may use similar or different tooling to achieve the removal of other components and materials. This is noted but does not form part of the scope of CARBOWASTE WP2, Tasks 2.1 to 2.3. The Reactor Dismantling Activity may cause some cross contamination of the i-graphite and this will be noted.

Retrieval and Segregation Toolbox Compartment
The toolbox is a term used in WP1 to reflect a vehicle for assisting in deriving integrated waste management solutions for i- graphite. The toolbox compartment will contain those specific tools that will assist users to consider how to retrieve and segregate i-graphite. Such specific tools will be referenced to the main tool box where necessary.

Route Map: An upper level analysis developed by Work Package 1 to link the anticipated inputs, issues, requirements and endpoints for all steps of the i-graphite integrated waste management approach.

Stakeholder: In the context of this report, any person, group of persons or organisation that will be impacted by the i-graphite retrieval and segregation process. Where organisations such as regulators, operators and contractor have already been implied the term will generally be applicable to the general public, other industries, local government, non-governmental organisations etc.

Unconditioned: State of waste as it is retrieved before it has been placed in its disposal or storage box and grouted or otherwise treated and immobilised. It may have been size reduced or segregated from other materials. Waste in this condition is not likely to be considered passive. This is generally the condition of i-graphite operational waste removed from reactors and stored in vaults or silos as was the case at Berkeley,

Hunterston A, Vandellos, some French UNGG and RBMK. That at Vandellos has been conditioned and packaged.

Toolbox Compartment: See Retrieval and Segregation Toolbox Compartment definition

Unit Operation: Term usually used in Chemical Engineering whereby specific steps of a process are specified and designed based on input and output and environmental conditions. So selection of a specific tool for say lifting a graphite block will have

Input: location, initial (condition, weight, dimension), associated components, number total and number per unit time to be removed.
Output: tool handling device, final (condition, weight, dimension), segregation, handling, packaging, number of pieces, number per unit time.
Environment: in-air, underwater, gas, temperature, pressure, radiation

All of these and any other relevant data should be specified as far as possible before the tool is selected or designed

Waste Disposal Route: is the route after ***dispatch*** that may lead to a variety of end-points for i-graphite depending on its radioactive content or subsequent processing or recycling activity; it may include the use or not of intermediate (as opposed to process buffer) storage.

Waste Packaging Efficiency: The term used to either refer to the concept or to measure the ratio between the final volume of waste as packaged as opposed to its volume before it is placed in a package. Waste Packaging Efficiency can be increased by adjusting the ratio of sizes of components placed in the package, by making better use of void spaces, by appropriate choice of package sizes, by volume reduction e.g. crushing, to name some options.

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APPENDIX 2 Glossary of Terms

AGR	Advanced Gas Cooled Reactor
AVR	Arbeitsgemeinschaft Versuchsreaktor (HTGR for research at FZJ)
ALARA (P)	As Low as Reasonably Achievable (Practicable) – note the term ALARP is used in the UK as it has been defined in legislation.
Berkeley	A twin Magnox steel RPV plant in SW England
BPEO	Best Practicable Environmental Option
BPM	Best Practicable Means
Bradwell	A twin Magnox steel RPV plant in SE England
Bugey	UNGG located at Bugey near Lyon in France
CARBOWASTE	Framework 7 Euratom Research Project that is the parent project for this report
Calder Hall	The first Magnox power station in the UK located at Sellafield commissioned in 1956.
CCTV	Closed Circuit Television
CFD	Computational Fluid Dynamics
DIDO	A high flux research reactor that contains a graphite reflector. Used a several sites including Harwell, UK and Jeulich, Germany
EdF	Electricité de France (French Utility)
EHS&Q	Environment, Health, Safety and Quality
ENRESA	Empresa Nacional de Residous Radiactivos, S.A. (Spain)
EPRI	Electric Power Research Institute
EU	European Union
Fort St Vrain	Location of a HTGR decommissioning project in USA that is a reference example of interest for decommissioning underwater
FI	Fizikos Institutas (Institute of Physics, Lithuania)
FZJ	Forchungszentrum Juelich GmbH (Germany)
GLEEP	Graphite Low Energy Experimental Pile (Located at Harwell, UK, now decommissioned)
HAZAN	Hazard Analysis
HAZOP	Hazard and Operability Study
Hinkley Point A	A twin Magnox steel RPV plant in SW England
HEPA	High Efficiency Particulate Absorbing (or Arrestance) applied usually to air filters
Heat Exchangers	In context of Magnox, AGR and UNGG plant are the primary cooling components for the reactor carbon dioxide cooling gas. Sometimes referred to as steam generators, sometimes referred to as boilers.
HLW	High Level Radioactive Waste
Hunterston A	A twin Magnox steel RPV plant in SW Scotland. Also has graphite sleeves stored in vaults
HTGR	High Temperature Gas Cooled Reactor

IAEA	International Atomic Energy Agency
Ignalina NPP	RBMK located near Visaginas in Lithuania
INR	Regia Autonoma Pentru Activiati Nucleare – Sucursala Cercetari Nucleare Pitesi (Institute of Nuclear Research, Romania)
LEI	Lithuanian Energy Institute
Latina	Location of a Magnox reactor in Italy
Leningrad NPP	A RBMK located close to St Petersburg in Russia which has previously removed graphite channels as part of a reactor upgrade.
LLW	Low level Radioactive Waste
Magnox	CO2 Gas-cooled Graphite Moderated Reactor used in UK, Italy and Japan
MERLIN	A swimming pool research reactor type supplied by Merlin Corporation.
MCDA	Multi-Criteria-Decision-Analysis
NNL	National Nuclear Laboratory (UK)
NPP	Nuclear Power Plant
Oldbury	A twin Magnox concrete RPV plant in SW England
RBMK	Reaktor Bolshoy Moshchnosti Kanalniy (High Power Channel Reactor (e.g. Ignalina NPP Lithuania)
POCO	Post operational clean out (e.g. emptying of vaults or silos of operational waste)
PPE	Personnel Protective Equipment
RPV	Reactor Pressure Vessel
SEPA	Scottish Environment Protection Agency
SINTER	The database accessible to EU research beneficiaries and used by CARBOWASTE to file documents and identification of participants. Information may be stored in private or public available areas. The access is http://w2ksrvx.ike.uni-stuttgart.de/sinter_neu/
SQEP	Suitably Qualified and Experienced Person
THTR	Thorium High Temperature Reactor
Tokai Mura	Location of a Magnox Reactor in Japan
UNGG	CO2 gas cooled graphite moderated reactor used in France (and similar in Spain)
Vandellos	Location of a reactor at Vandellos in Spain fairly similar to an UNGG. Already has had graphite operational wastes removed from Silos and stored on site.
WAGR	Windscale AGR
WP	Work Package (one of 6 study activities of the CARBOWASTE project)
Wylfa	A twin Magnox concrete RPV reactor in NW Wales. Differentiated from other UK reactors by a dry fuel route and is the last in the line of UK Magnox stations. Still in operation.

APPENDIX 3 Key Issues Affecting Retrieval and Segregation from Magnox, RBMK and UNGG Reactors

This appendix is issued as three separate attachments

These presentations are included in the notes of the workshop, which may be found on SINTER. CARBOWASTE document no: CARBOWASTE-0907-D2.2.1.

1 Presentation by M J Grave, Doosan Babcock at Workshop on 14th July 2009 on Magnox

2 Presentation by F Berenger, EDF at Workshop on 14th July 2009 on Bugey / UNGG decommissioning

3 Presentation by P Poskas, LEI at Workshop on 14th July 2009 on Retrieval of Graphite from Ignalina NPP

PRESENTATION BY

M J Grave, Doosan Babcock

Magnox (Oldbury & Hinkley Pt A)

Date 14 July 2009 at Doosan Babcock, Gateshead

Relevant Background Data

- WP1 Review Report (1st draft Eccles, Metcalfe & Banford)
- Doosan Babcock internal studies on Magnox Decommissioning
- Hinkley Point A Graphite Core Descriptions
- Oldbury Graphite Core Description
- Wylfa Anti Lift devices (awaited)
- All Magnox designs different
- Hinkley and Oldbury representative
  - Hinkley Spherical Steel RPV and zirconium pins
  - Oldbury Concrete RPV and graphite keys

Contributory Impacts

- “In a reactor the only property of graphite that does not change is the colour”

Irradiation Damage in Graphite, B T Kelly, J E Brocklehurst and JH Gittus, 1972

- Physical / Chemical Properties of i-graphite and other associated materials
  - Dimensional changes
    - *Isotropic graphite will shrink in all dimensions under all irradiation conditions.*
    - *Anisotropic graphite will shrink in all dimensions under irradiation at temperatures greater than 300°C.*
    - *At lower temperatures, anisotropic graphite will shrink in one direction and expand in the perpendicular direction.*

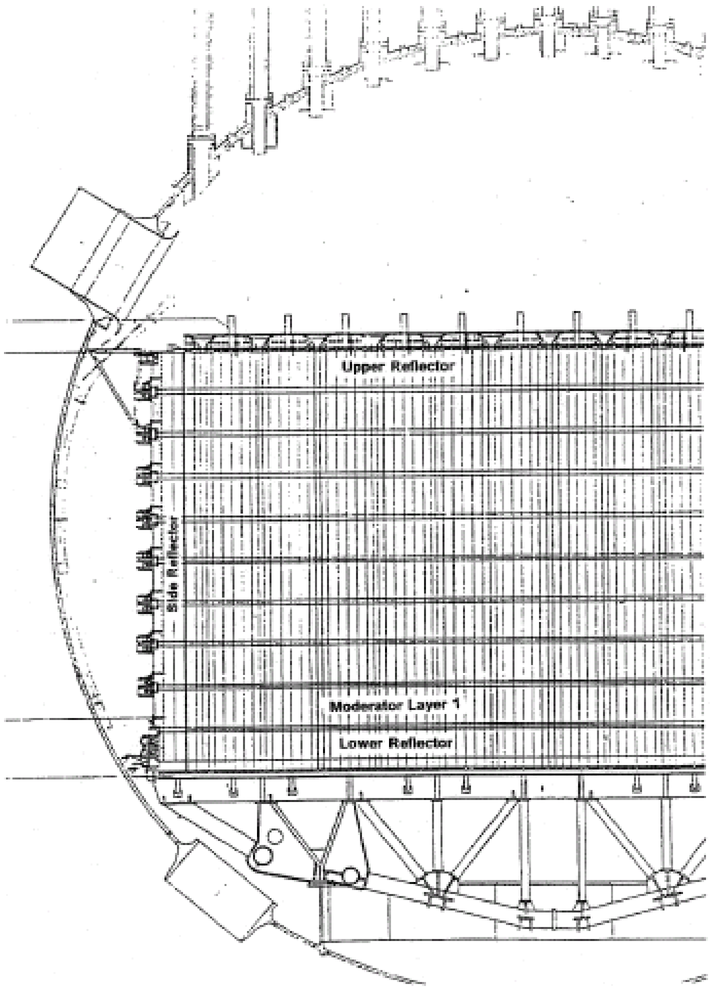
Contributory Impacts

- Modulus and Strength
  - *The modulus of graphite irradiated at small and medium dose levels will increase. This is accompanied by an increase in strength.*
  - *At high dose levels the graphite structure breaks down resulting in the modulus dropping drastically.*
- Oxidation of Graphite
  - *In air, significant thermal oxidation of graphite will not occur below 350°C.*
  - *During operation of CO2 cooled reactors, radiolytic oxidation of graphite by carbon dioxide produces carbon monoxide. Radiolysis of carbon monoxide produces carbonaceous deposits on cooler graphite surfaces.*
- Graphite dust explosion not considered possible.

Potential Methods for Removal

- **Windscale AGR**
  - Is an AGR not a Magnox
  - Is a smaller core (but RPV is deep and narrow)
  - Graphite has been completely removed
  - Described in WP1 Summary Report and presentation to WP2 Manchester meeting
- **Magnox**
  - UK Magnox plant and Latina (studied by UK engineers) studied since 1980s
  - Tokai study reported in Japan chapter covers multiple-grab for graphite
  - Considerable in service inspection and sampling of i-graphite
  - No decommissioning activity
  - Some if not all Magnox reactors will fall in to the 75 year category
  - Focus on two “most representative” Magnox designs

Hinkley Point A General Description



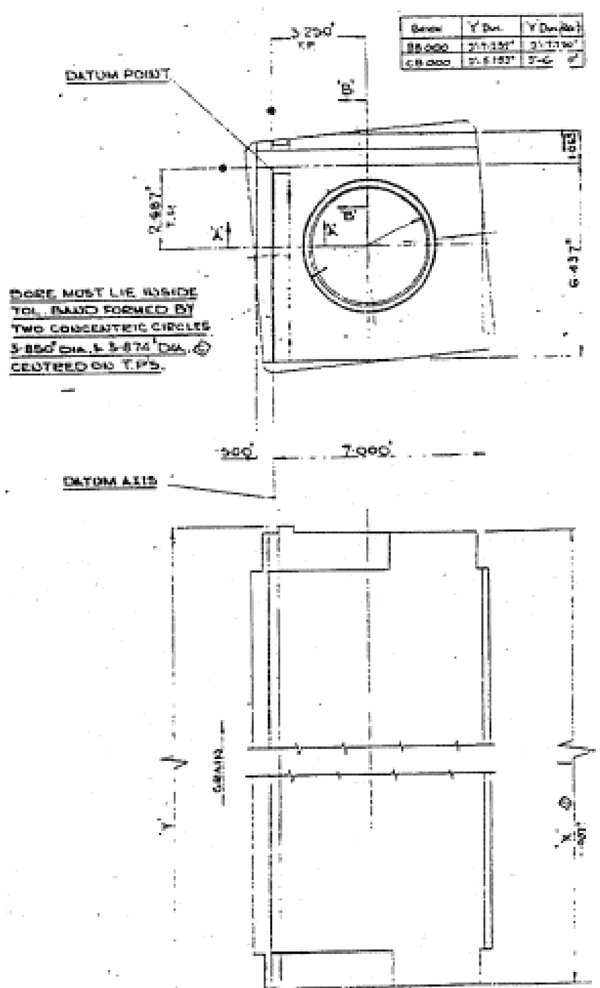
Reactor Layout

- Approx Core Dimensions
  - Height: 8.8m
  - Diameter: 16.2m
- Divided into columns of bricks which are mostly centrally bored
- Each column consists of:
  - 10 layers of bricks separated by:
  - 9 layers of tiles
- Graphite components:
  - Bricks
  - Tiles

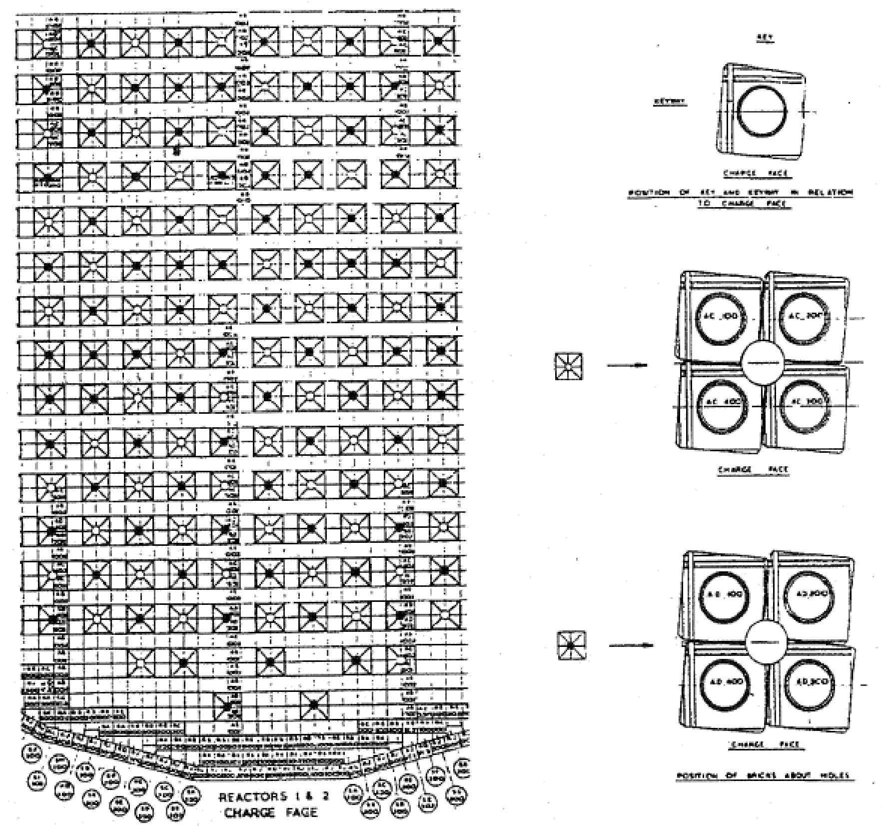
Hinkley Point A General Description

- Zirconium alloy pins used to locate neighbouring bricks and tiles
- Successive brick layers rotated 2 deg in alternative directions with respect to the vertical axis to prevent neutron streaming
- Core restrained by 10 garters which intersect with each tile layer and the lower and upper reflector layers
- Core monitored by:
  - 204 single thermocouples
  - 30 triple thermocouples
- Magnox gas seals between brick/tile junctions

Hinkley Point A General Description

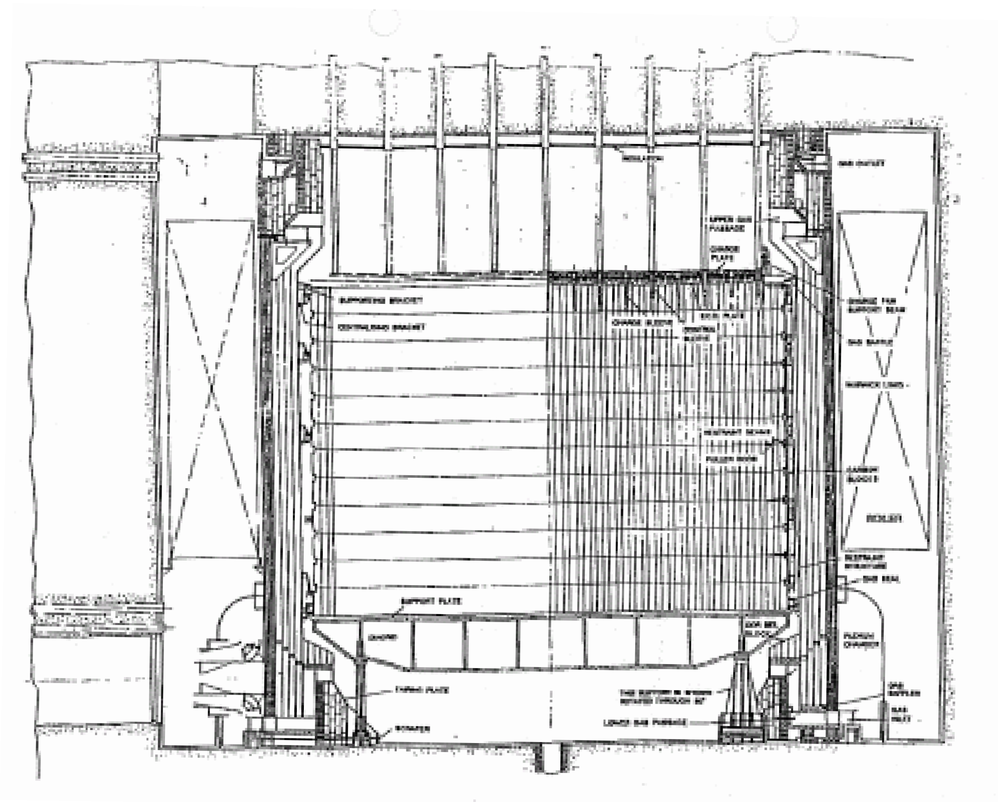


Brick Design



Core Layout

Oldbury General Description



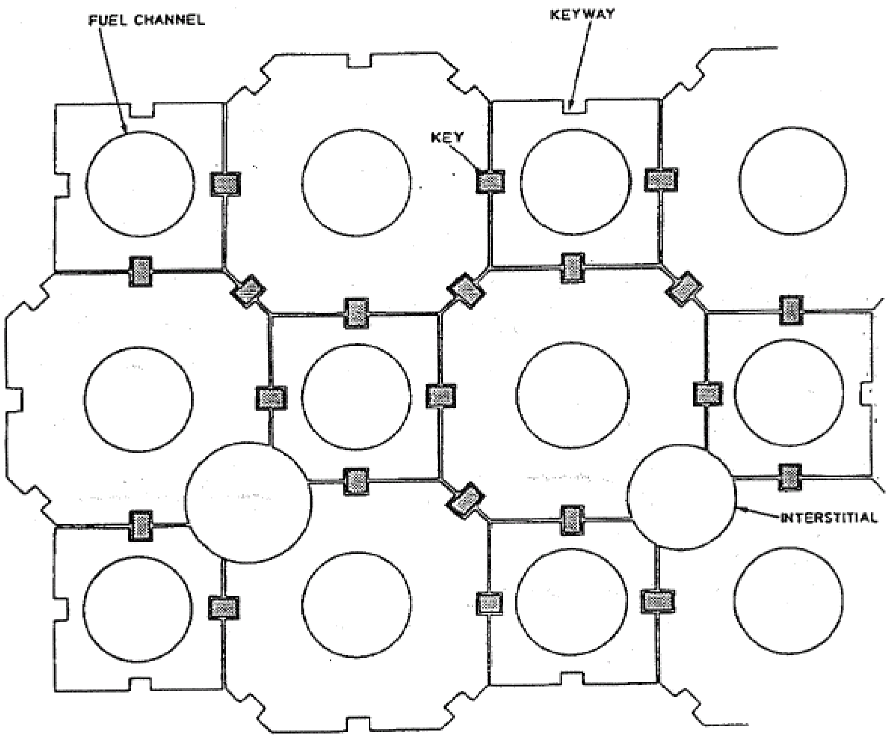
Reactor Layout

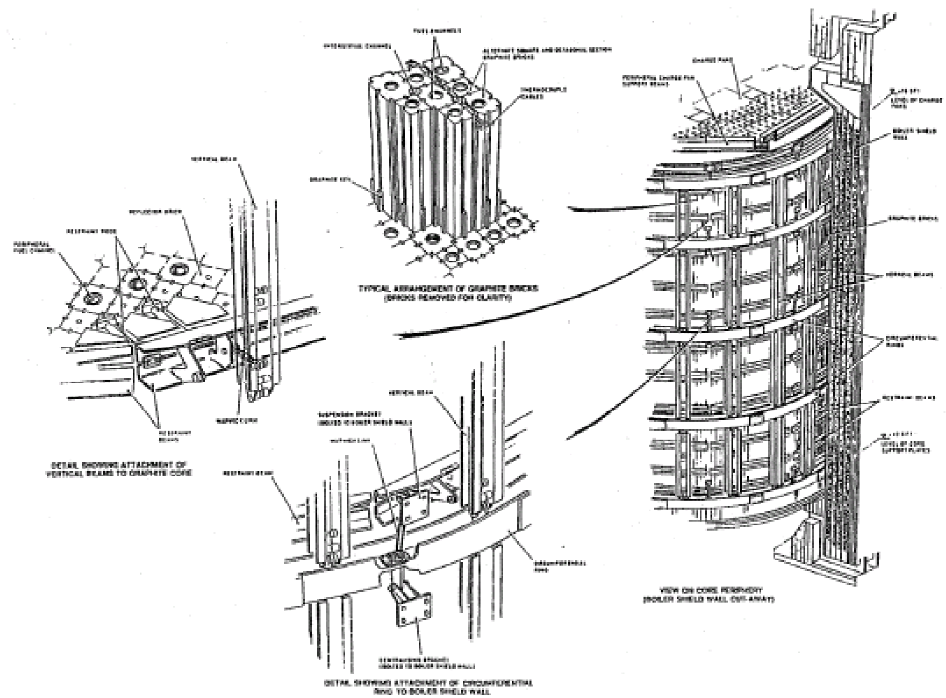
- Approx Core Dimensions
  - Height: 9.8m
  - Diameter: 14.2m
- Divided into columns of bricks which are centrally bored
- Each column consists of 12 layers of bricks
- Graphite components:
  - Bricks
  - Keys

Oldbury General Description

- Two different types of brick
  - Square
  - Octagonal
- Graphite keys used to locate neighbouring bricks
- Core restrained by 5 circumferential rings connected by vertical beams
- Core monitored by 325 thermocouples
- Magnox gas seal between brick junctions

Oldbury General Description





Core Layout

Restraint System

Comparison of Representative Stations

	Hinkley Point A	Oldbury
Core Shape	Vertical Cylinder	Vertical Cylinder
Core Dimensions	Diameter: 14.2m Height: 9.8m	Diameter: 16.2m Height: 8.8m
Core Weight	2210 tonnes	2061 tonnes
Method of Location of Bricks	Zirconium Pins	Graphite Keys
Brick Shape	Square	Square and octagonal
Pressure Vessel	Steel Sphere	Concrete Domed Cylinder

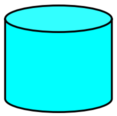
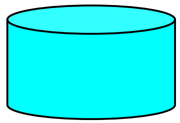
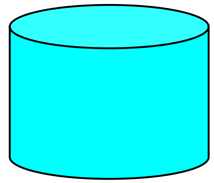
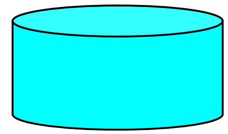
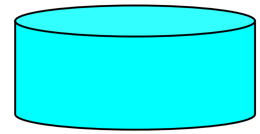
Advance to a Commercial Solution

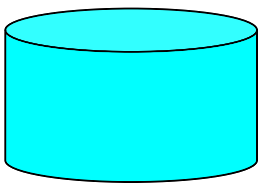
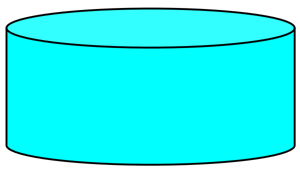
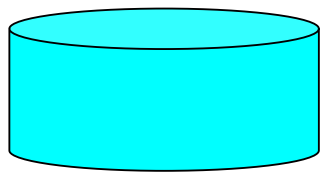
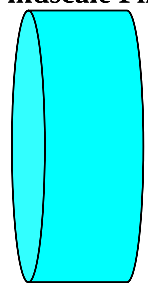
- *Decision by NDA to fund*
- *Definition of Waste Route*
- *Windscale AGR used*
  - Letter of compliance from Nirex (now NDA)
  - Availability of Interim Waste Store
  - Contractors designed and built a preliminary waste route adjacent to the reactor which input to non-standard waste disposal box and grouting facility
  - Some modifications to reactor area to fit in waste route (jack up boilers)
- *For Magnox range of graphite removal dates could range from less than 25 to 75*
- *ILW repository not expected until 2040*
- *Both WAGR and GLEEP demonstrate that graphite can be removed using commercial technologies provided a preliminary waste route is available.*
- *Underwater for Magnox not considered so far but Windscale Pile was proposed at tender stage.*

Decision Support Data

- Modelling of graphite in UK exists but largely aimed at reactor operational support. Graphite is inspected and tested.
- Physical modelling of processes (see works visit and WAGR tested processes)
- Regulator comment on i-graphite management
    - Non-specific
- Cost data available e.g. budgets
    - Non specific in public domain (NDA LTP includes reactor Decom)
- NB this is work in progress

Core Size

Windscale AGR	Calderhall/Chapelcross Magnox	Hartlepool AGR	Ignalina RBMK	Hunterston A Magnox
				
Height: 6m Diameter: 6m Weight: 285 tonnes	Height: 6.4m Diameter: 9.45m Weight: 1164 tonnes	Height: 9.8m Diameter: 11.2m Weight: 1360 tonnes	Height: 7m Diameter: 11.8m Weight: 1700 tonnes	Height: 7m Diameter: 13.5m Weight: 1780 tonnes

Oldbury Magnox	Hinkley A Magnox	Wylfa Magnox	Windscale Piles
			
Height: 9.8m Diameter: 14.2m Weight: 2090 tonnes	Height: 8.8m Diameter: 16.2m Weight: 2210 tonnes	Height: 9.14m Diameter: 17.4m Weight: 3470 tonnes	Depth: 7.4m Diameter: 15.3m Weight: 2000 tonnes



Carbowaste WP2

14-16/07/2009

Newcastle upon Tyne

F. BERENGER EDF

Summary

Bugey 1 decommissing flow diagram complement

Evaluation phase addition

- Technical design detail
- Safety issues
- Risk assessment

Scenarios choice matrix

Bugey 1 core brick retrieval risk analysis

Flow diagram complement

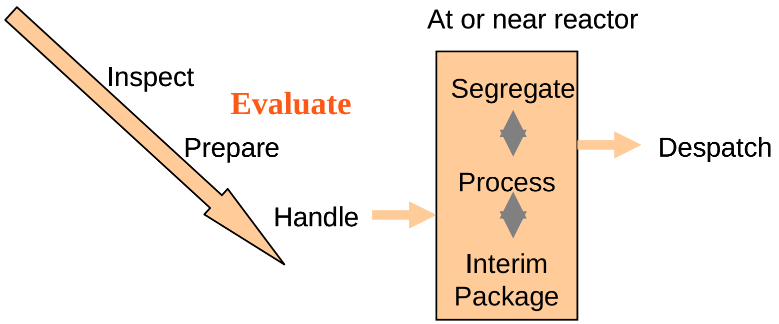
WP2 flow diagram completion with Evaluation

Reactor design main characteristics and its effects on scenarios predictions

Decommissionning safety assement

Decommissionning efficiency

- Level of risk against investment to reduce it
- Delay
- Cost
- Technical challenges



Flow diagram complement

Evaluation process for Bugey 1 overall decommissionning process :

Reactor vessel is vertical and can be flooded so underwater decommissionning induce :

1. **water contact trap lot of contamination :**
   - safety cases assess reduced gaz discharges levels
2. **irradiation protection**
   - reduce decommissionning dose
3. **visibility of workzone**
   - reduce delay
   - reduce level of risk caused by uncertainities

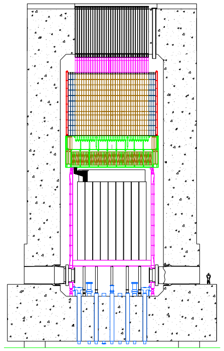
Scenario choice matrix

Choice Matrix to be completed

high dose rate area

transparent protection | or | remote operation

Underwater | | robotised tools + vision equip.



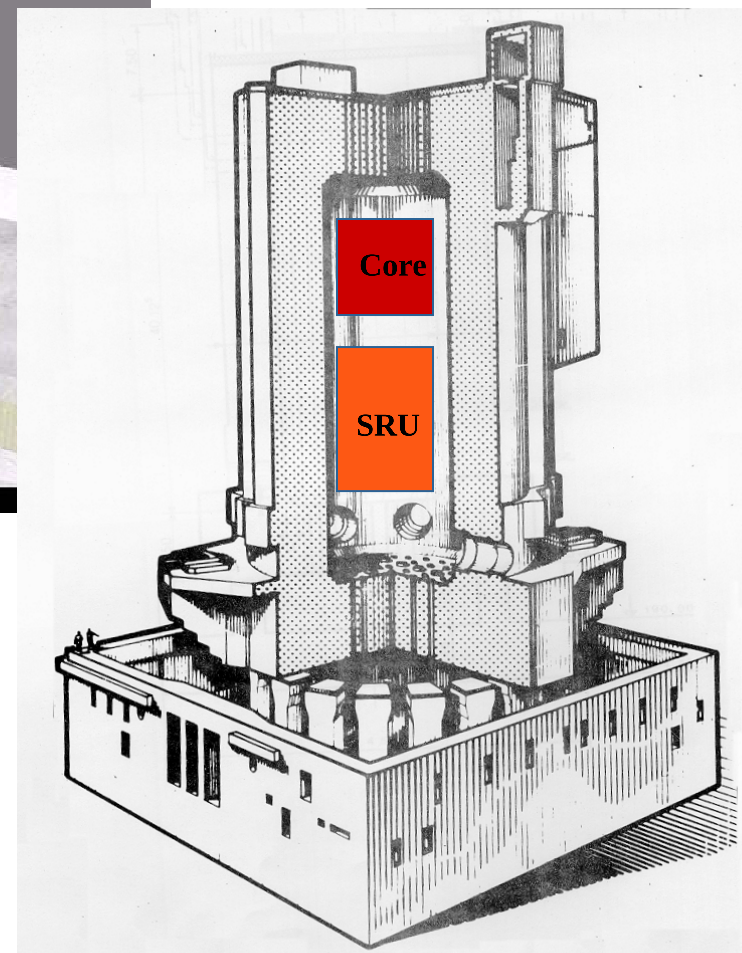
Vertical containment | | All arrangement

with high component density | | low component density

Graphite equipement decommissionning must be considered as a part a global decommissioning operation : Tools developped to cut and handle metallic support structure could be adapted for graphite parts handling graphite is not a critical activity

Retrieval risk analysis

Bugey 1



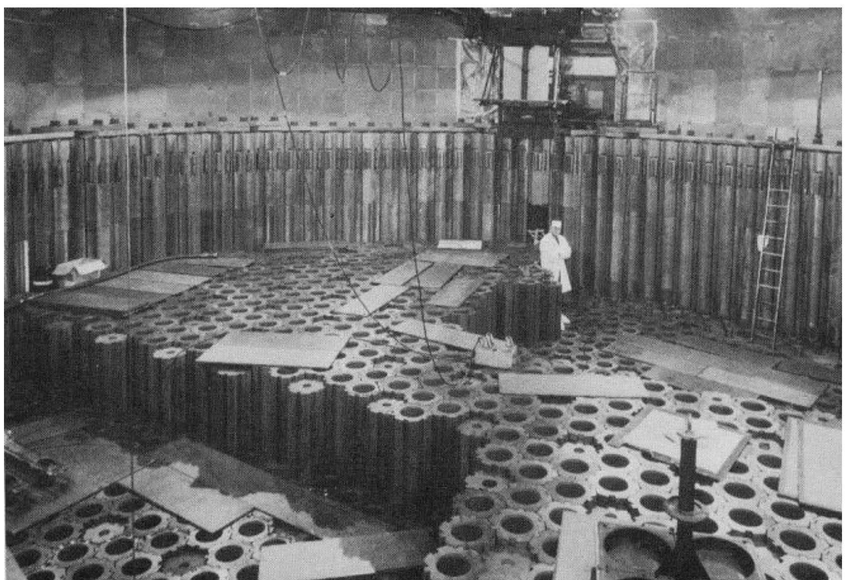
Retrieval risk analysis

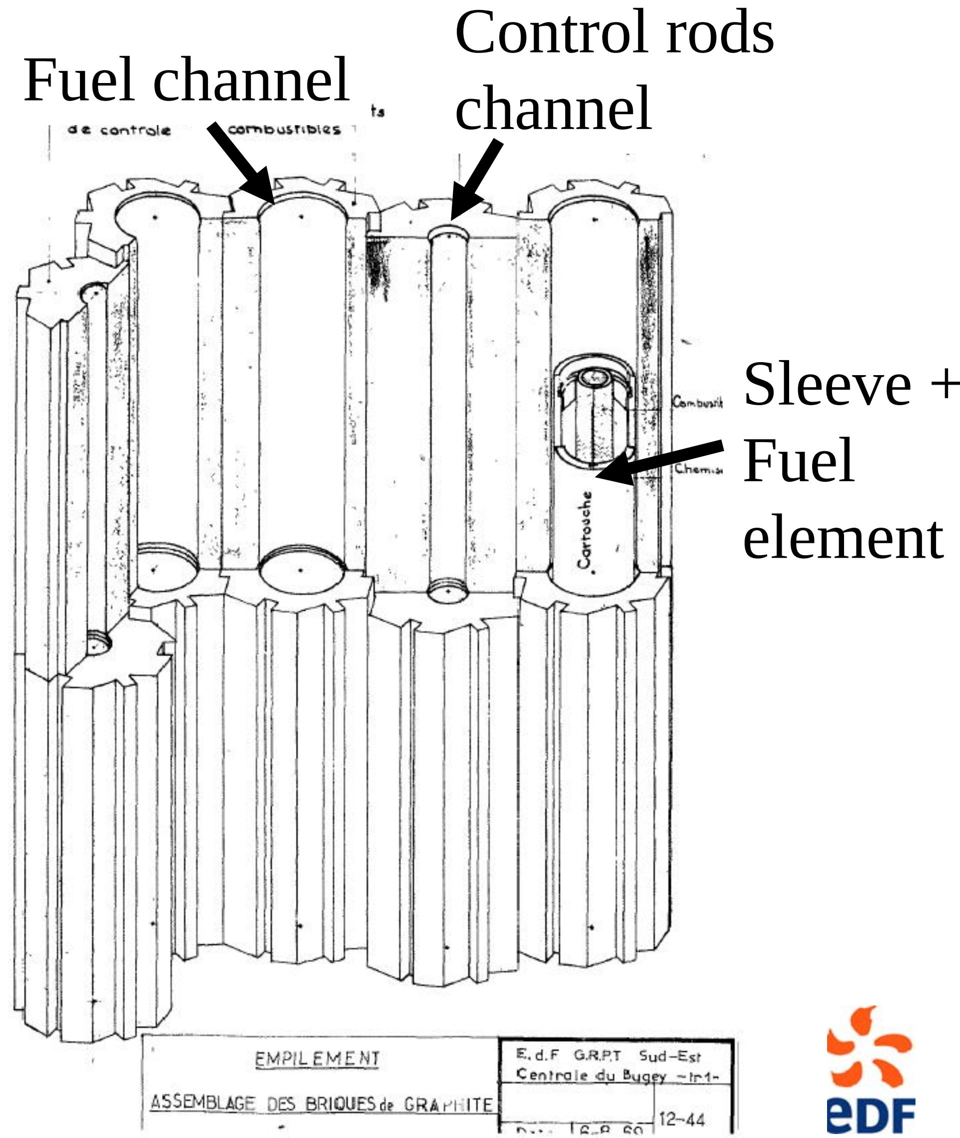
Bugey 1 core

852 fuel channel

~2000 t virgin G

~15 m diameter / 10 m height





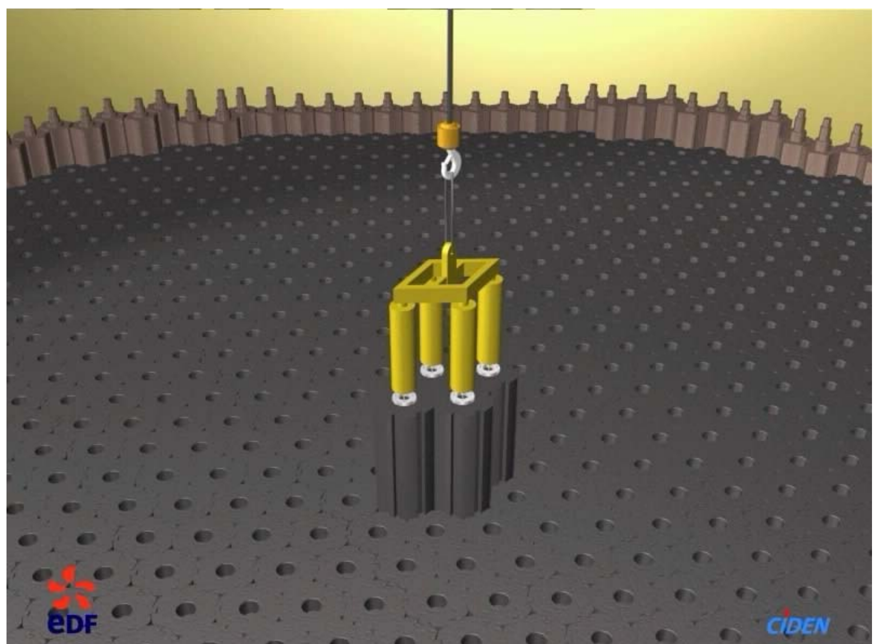
Retrieval risk analysis

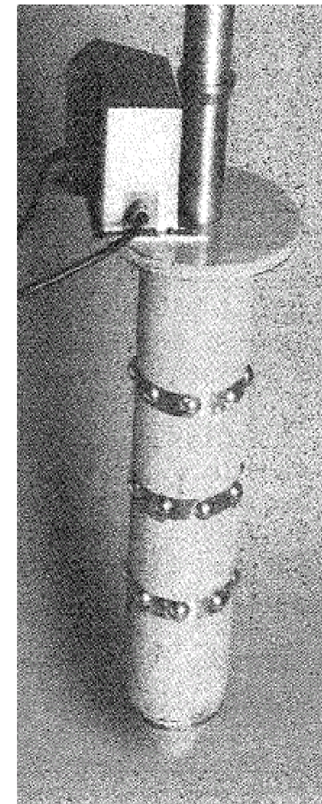
Bugey 1 decommissionning scenario :

Reactor containement water flooded

Brick handling : alone or by group of 2 or 4

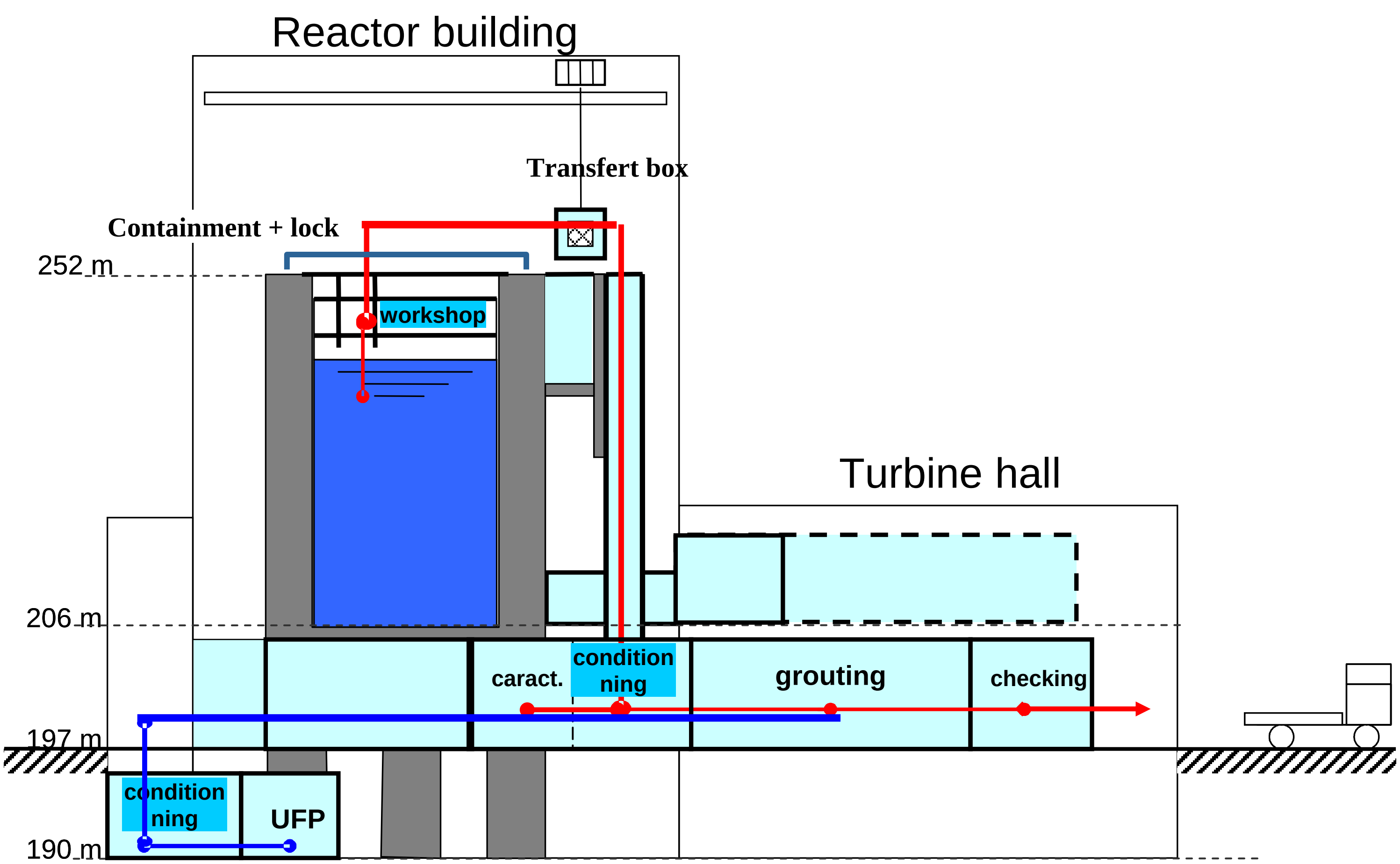
Tools : ball grabs like in WAGR





Retrieval risk analysis

Bugey 1 : graphite and under water induced waste route



Retrieval risk analysis

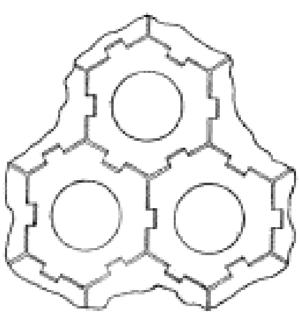
Bugey 1 : graphite core design details :

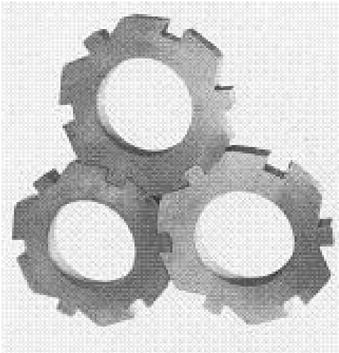
Graphite evolution under neutron fluxes in CO2 cooled reactors where well known when Bugey 1 stack was designed :

- Wigner dimensionnal, thermal changes
- CO2 weight losses, methane protection

As the deposited specific energy reach high values, strong dimensionnal changes where avaited and a very good link between brick where found in Tokaï Mura 1 brick linckage system :

TOKAI MURA





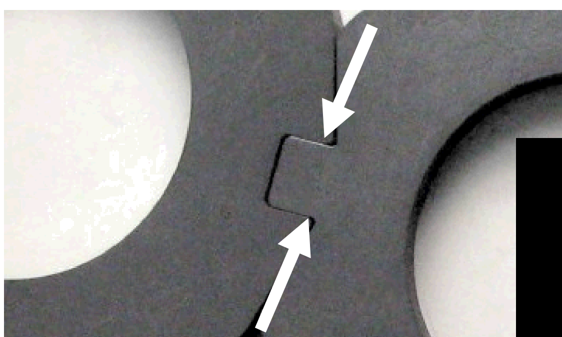
Bugey 1

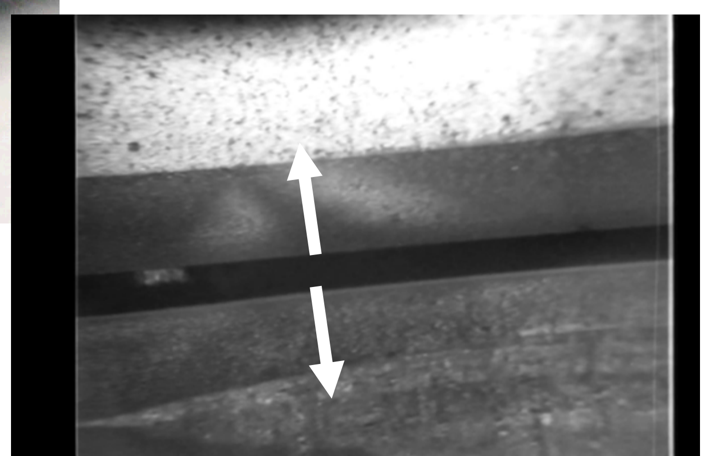
Retrieval risk analysis

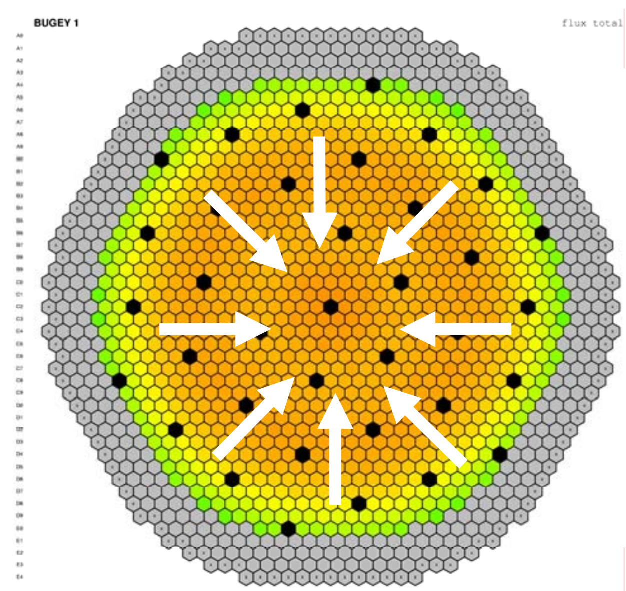
Bugey 1 : graphite core state knowledge

Qualitative data compilation give raise to the following states :

- Brick internal stresses are high
- Mecanical strenght is reduced by weight losses
- Thermal gradient induced tenon-mortice arrangement clamping
- Long distance cascade effect clamps collumns of bricks together







Retrieval risk analysis

Bugey 1 : graphite core state knowledge

Global core modelisation look not promising since hight uncertainities exist on main sensitives parameters :

- Temperature gradient evolution caused by brick to brick cool gaz leakage,
- Graphite dimensionnal evolution linked with porosity evolution,
- etc

Direct measure of internal stress level look promising but is a big challenge since lots of sensitives parameters have to be estimated before stress evaluation :

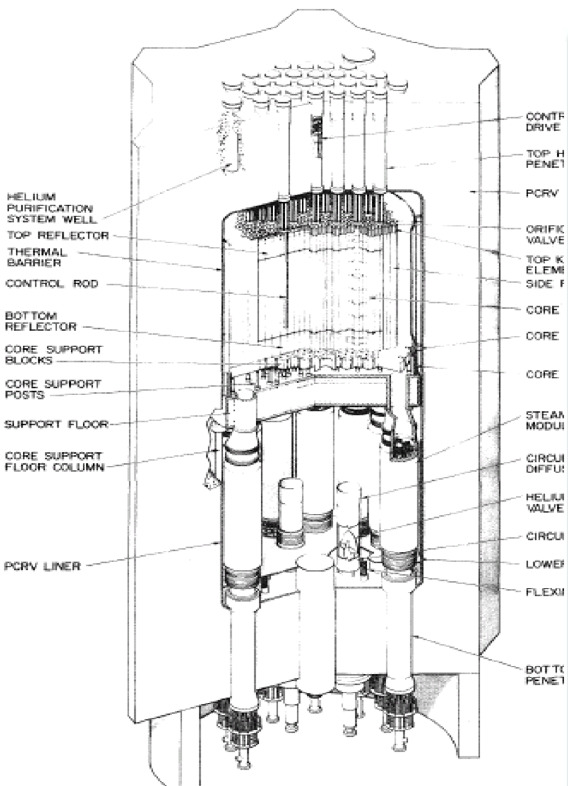
Apparent Young’s modulus, length, unstressed blanck,…

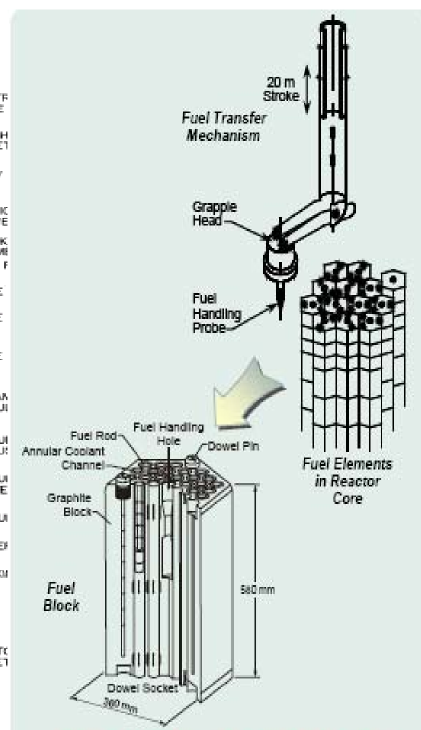
Linked with the high heterogeneity of the graphite matrix cuased by non uniform methane protection.

Retrieval risk analysis

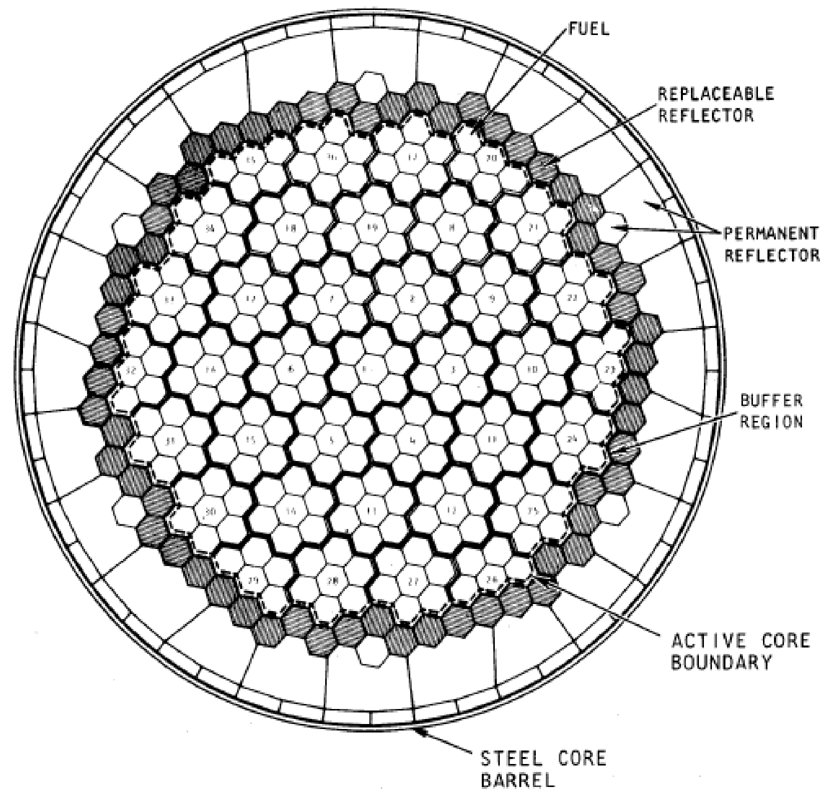
Fort St Vrain Experience :

Core internal graphite blocks unloaded by fuel handling machine, only reflector 1m thick to decomission.





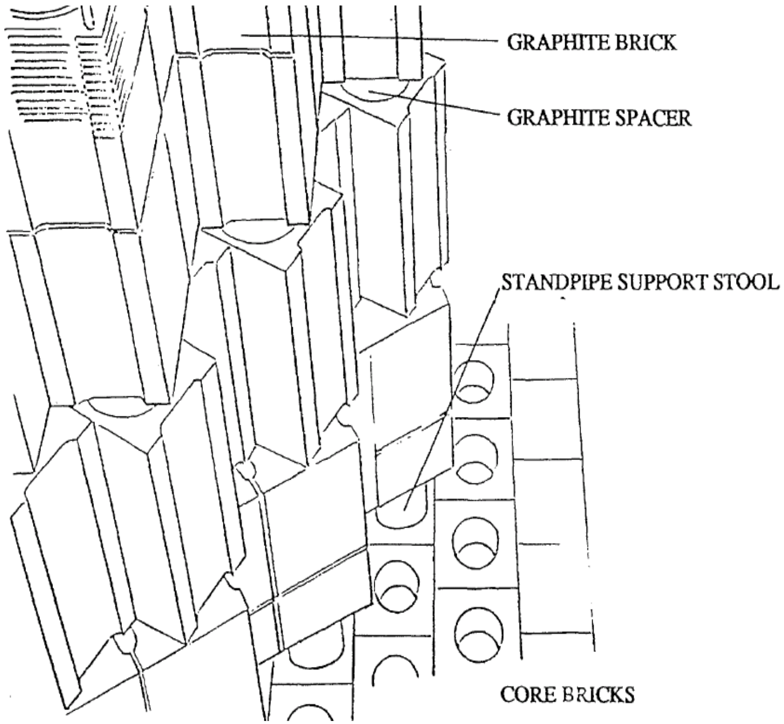
Prismatic Fuel Block

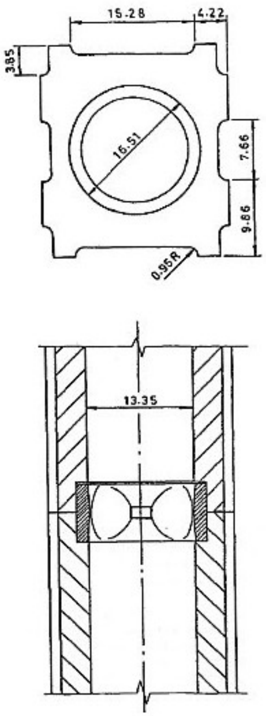


Retrieval risk analysis

WAGR Experience :

Scare bricks no lateral link, operationnal schrinkage -> enance clearance, just bricks to handle



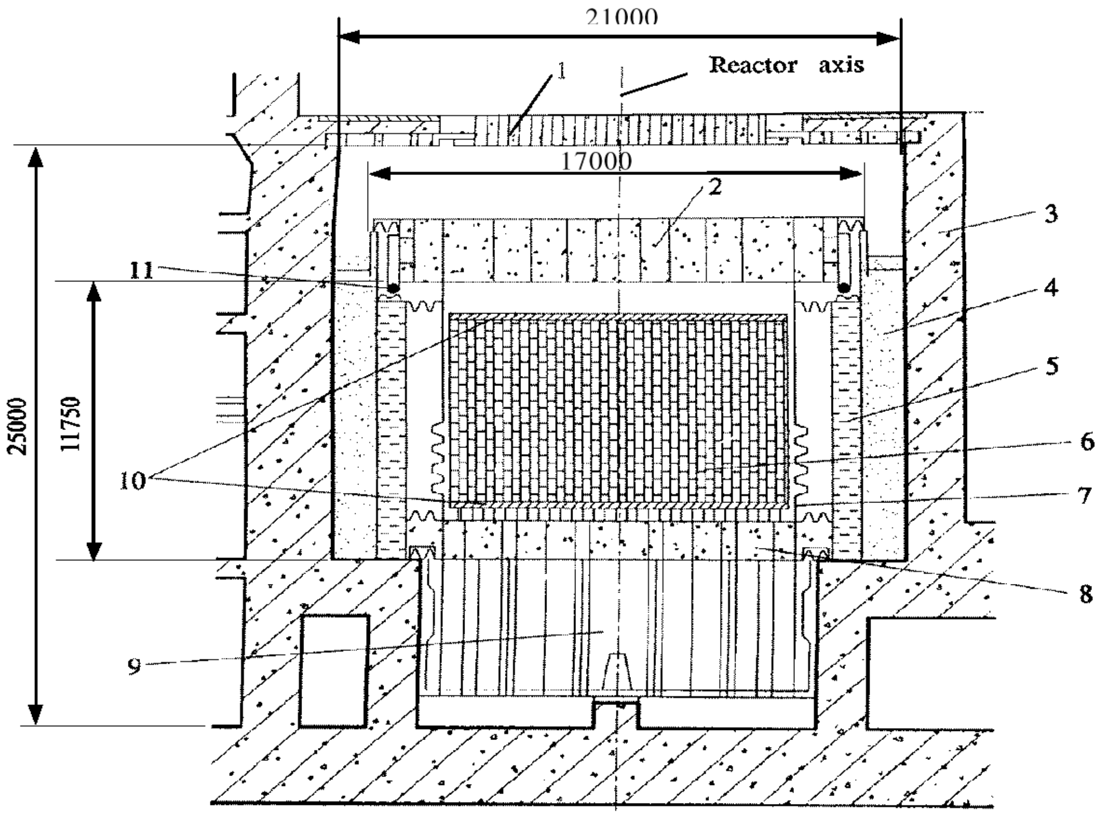


- PRESENTATION BY
  - *Pavilas Poskas, LEI*

- TITLE
  - *RBMK (Ignalina)*

- Date 14 July 2009 at Doosan Babcock, Gateshead

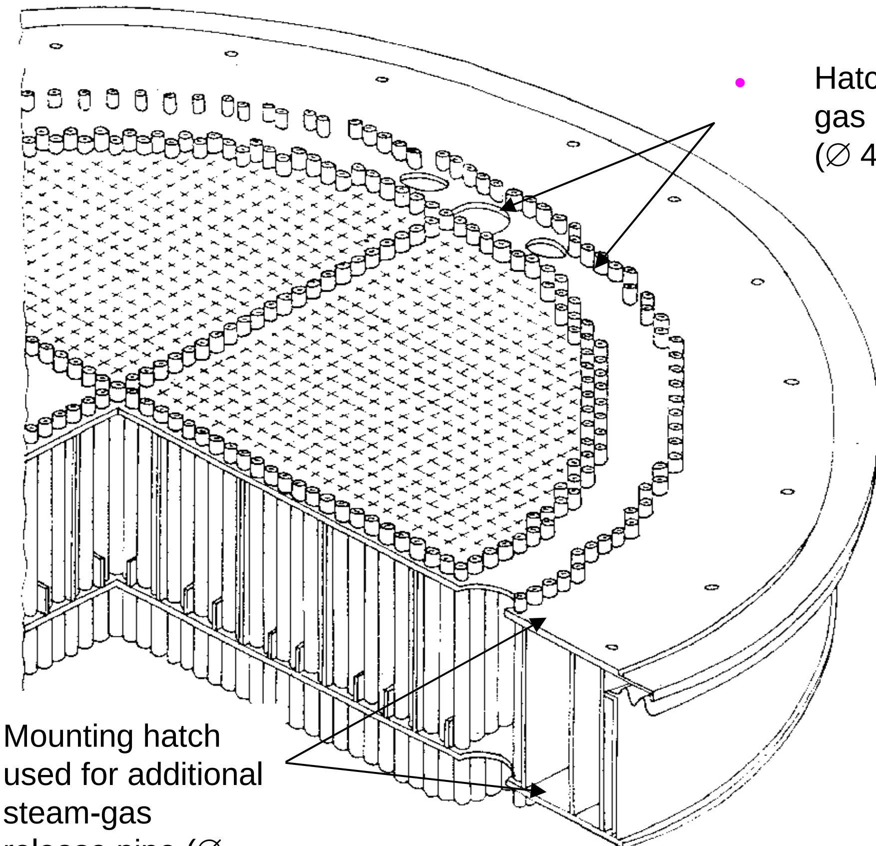
Relevant Background Data



RBMK-1500 reactor

1 – plate floor, 2 – top metal structure with serpentinite filling, 3 – concrete vault, 4 – mounting space with sand filling, 5 – lateral biological shield (annular water tank), 6 – graphite stack, 7 – reactor case, 8 – bottom metal structure with filling, 9 – reactor support structure, 10 – support (bottom) and shield (top) steel plates, 11 – roller supports

Relevant Background Data

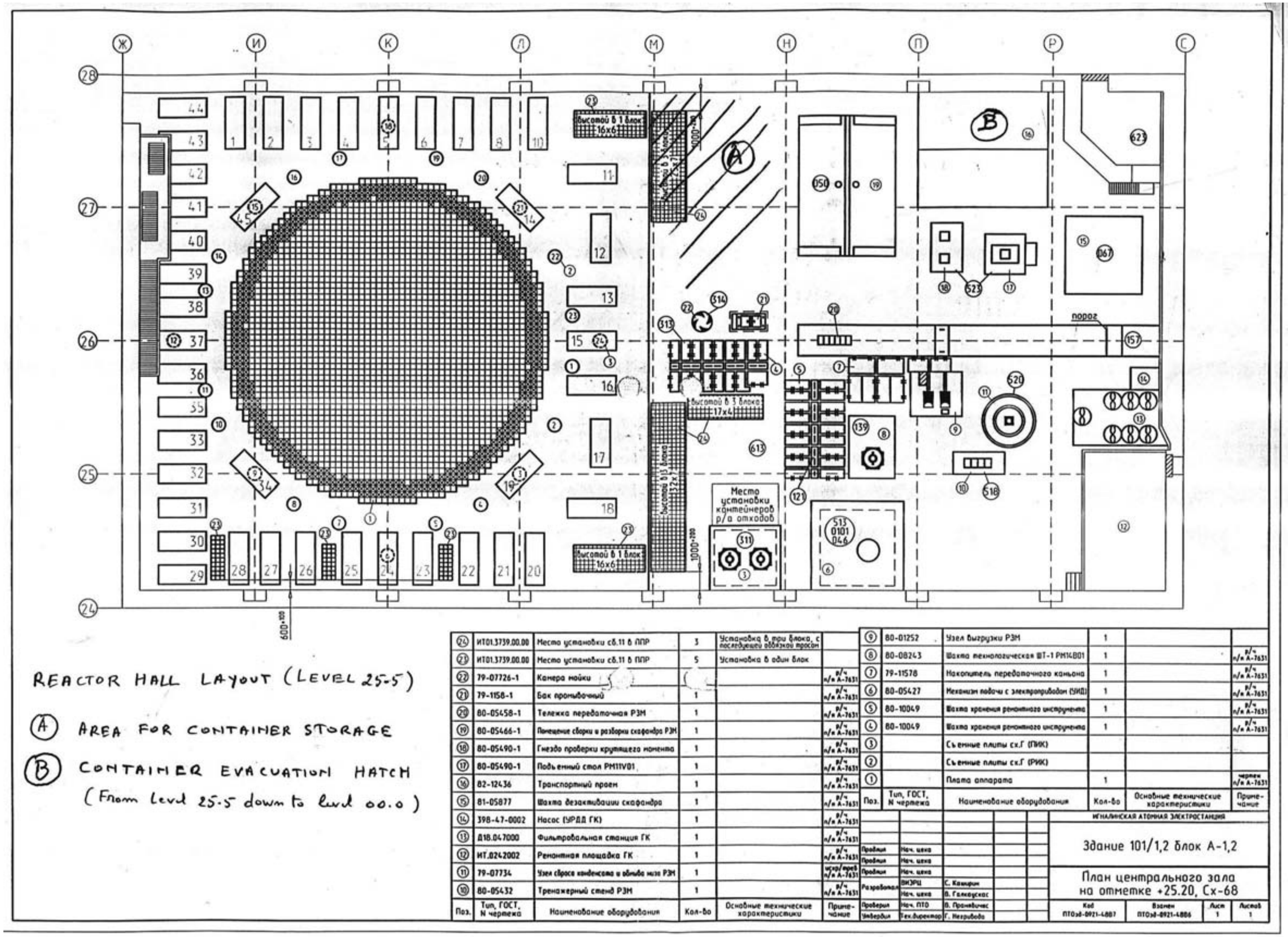


RBMK-1500 reactor

Top metal structure

- Overall dimensions of structure, mm Ø 17650x3000.
- Mass of metal structure 593405 kg.
- Mass of filling, not less, 1005000 kg.

Relevant Background Data



Plan of Reactor Hall

Relevant Background Data

- Total amount of channels in reactor core is 2070, including:
- · fuel 1661;
- · CPS (reactor control and protection system) 235;
- · reflector cooling 156;
- · temperature 17;
- · gas sampling 1.
- Core dimensions: Ø 11800 mm, height - 7000 mm. Core lattice is square with pitch 250 mm.
- 17 temperature channels and gas sampling channel are located outside the lattice.

Relevant Background Data

- Brief reference to study work already undertaken
  - Review reports, Data Pack, Route Map Step 2
- Focus on what this workshop needs to know (if no info, say so!)
  - Contributory impacts
    - *Comment on issues on Route Map Step 2 see – next slide*
  - Major issues
    - *Indicate state of development for i-graphite retrieval*
    - *What is needed to advance to a commercial solution*

Comment on Contributory Impacts

- Physical and Chemical Properties of graphite
  - Efficiency of the graphite stack ("Standards for Calculation of Strength of Graphite Assemblies and Graphite-uranium Reactor Parts" named НГР-01-85 in Russian, Ref. No. К-3624, refer) is determined by the following criteria:
    - · *Critical graphite damage level (neutron critical fluence, Fcr), under which a fast degradation of graphite properties happens, leading to graphite, as structural material, loss of bearing ability;*
    - · *Critical change of graphite units (bricks) geometry, under which the reactor core efficiency disrupts in the whole – exhaustion of technological radial expansion gap in the system: graphite units (bricks) – graphite rings*(sleeve) – fuel channel (jamming of FC in the graphite stack and possible destruction of graphite components when FC is being extracted);
    - · Graphite blocks integrity failure (intense cracking), under which abruptly disrupts the blocks' ability to function as the core area structure components and which requires a special attention during reactor dismantling.

Comment on Contributory Impacts

- Physical and Chemical Properties of graphite
  - In accordance with the special "Regulations for the In-Service Inspection of Fuel Channels, CPS Channels and Graphite Stack of the RBMK-1500 Reactor" there is a periodical monitoring of the graphite stack provided, which includes a visual graphite check with a television camera after FC is extracted in order to detect mechanical damages (cracks, chips, holes and etc.).
  - The criterion of neutron irradiation critical fluence, neutron fluence, is used as radiation resistance criterion for graphite items (graphite units (bricks), solid contact rings (sleeves) and etc.).
  - When it is reached, linear dimensions of the graphite items after shrinkage stage reach reference value.
  - When this neutron fluence is reached, graphite is in the "secondary swelling" stage, which is characterized by intense cracking, rapid deterioration of all graphite physical and mechanical properties (strength decrease, heat conduction and etc.). This leads to its efficiency loss.

Comment on Contributory Impacts

- Physical and Chemical Properties of graphite
  - According to the "Work programme on arrangement of reference graphite bushings in temperature channels of enterprise Unit 1, A-1513 (INPP) and further control of graphite stack condition" the reference graphite bushings are installed in six temperature channels, cells 06,5-24,5; 22,5-24,5; 24,5-30,5; 26,5-24,5; 34,5-24,5; 42,5-24,5, as samples-evidences under reactor dismantling (Protocol ПНР-53 dated 1984.01.27).

Comment on Contributory Impacts

- Physical and Chemical Properties of graphite
  - In order to carry out postreactor research and efficiency forecasting duringplanned maintenance of 1992, there were graphite bushings extracted from the cell 06,5-24,5.
  - The reference graphite bushings have been cut out parallel and perpendicular to the unit axis (two cutting directions considering texture) from the ready-made graphite blocks of standard ГР-280 (in Russian) brand reactor graphite, which are installed in the INPP Unit 1 reactor core.
  - Metrological data analysis showed that virtually all bushings have been irradiated under temperature of (650÷700)°C to neutron fluence (3,1÷3,5)·1022 n/cm2.
  - Results showed that the graphite compression strength has increased by (63÷95)%, and coefficient of elasticity – by (51÷95)%.

Comment on Contributory Impacts

- Potential methods of removal
- Graphite Moderated Gas Cooled Reactors – RBMK Operational Maintenance and Repair Experience
- Due to the basic design differences from LWR, most of the today industrial decommissioning feed back is expected to come from the already performed and on-going graphite moderated and gas cooled decommissioning projects, at least for the preliminary dismantling phase (graphite removal and handling of pre-cut of components), as those reactors exhibit similar technological features with RBMKs.
  - Fort St Vrain (USA),
  - Windscale AGR and Magnox Reactors in UK,
  - Japan,
  - Italy,
  - HTR and AVR (Germany)

Comment on Contributory Impacts

- Potential methods of removal
- Final cut operations (size reduction) can be performed under water, for example in some of the emptied spent fuel pools, with already proven tools.
- Dismantling, technique and tool inputs are also expected to come from the RBMK Russian Designer, who has now tackled up the RBMKs dismantling studies.

Comment on Contributory Impacts

- Potential methods of removal
- However, as shown hereafter, the already accumulated experience during operational maintenance and repair works is of primary importance in preparing RBMK dismantling sequences.
- The following table compares the main technological features of INPP RBMKs and of the Graphite Moderated Gas Cooled Reactor.

Comparison of RBMKs and Graphite Moderated Gas Cooled Reactors Technological Features

Feature	Ignalina RBMKs	Typical Gas Cooled Reactor
Long channel components	Fuel and cooling water channels, control rods etc, below standpipes	Details vary, but basically channels for fuel, cooling gas and control rods below standpipes
Large area at charge face	Access to channels is through removable shield blocks at vault floor	Access to channels is through removable shield blocks at charge face
Above core structures	Top cover, steam/water Pipework bundles	Details vary, but basically, guide tubes, neutron shield, and pipe bundles
Graphite stack	Graphite moderator and reflector comprising several thousand blocks	Graphite moderator and reflector comprising several thousand blocks
Core support structure	Core support structure for graphite stack and other components	Core support structure for graphite stack and other components
Cylindrical structures	Diaphragm, water cylinder and sand cylinder around graphite	Core restraint structure around graphite. Not a continuous structure, but still requiring 360 degree access
Pressure Vessel	Doesn't exists, pressure is contained in the channels, the major key design difference, may or may not affect dismantling, according to option chosen etc.	Spherical and cylindrical pressure vessels, either steel or prestressed concrete
Concrete structures	Reactor is surrounded by the concrete vault	Reactor is surrounded by the concrete bioshield, either separately in the case of steel vessels, or integrally for prestressed concrete vessels
Penetrations	Large openings through the concrete vault for steam and water pipes, standpipes for fuel, control, and instrumentation	Penetrations through the bioshield for gas cooling, inlet and outlet, standpipes for fuel, control instrumentation

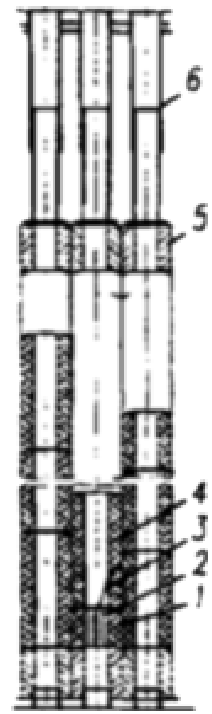
Comment on Contributory Impacts

- Potential methods of removal
- Leningrad nuclear power plant were tasked with solving the following problem:
- - to develop a technology that would make it possible to extract from the reactor an entire graphite column for shipment to IRTM of the Russian Science Center "Kurchatov Institute" and
- - to restore the graphite masonry of the reactor in the third power-generating unit, in which cell 52-16 operated without a graphite column 'after the liquidation of the consequences of a loss of seal in the access channel in March 1992.

Comment on Contributory Impacts

Leningrad NPP

- In 1992, during the liquidation of the consequences of the loss of seal in the access channel, the top steel plates of cells 52-16, 52-17, and 53-16 of assembly 07 were lifted into the extreme top position and clamped to the pipes in the top sections.
- All 14 graphite blocks were removed from cell 52-16 by grinding.
- In order to remove the fragments of graphite and fuel assemblies from the gaps, the adjoining columns were alternately moved, block by block, into the empty cell 52-16.
- Next, these columns were put back into their intended position. However, one 200 mm high block (position 2) from cell 53-16 and one 600 mm high block from cell 52-17 (position 4) where damaged when the blocks were moved.
- The first block had several cracks (position 1) over its entire height, and the second block had a 250 mm long U-shaped crack (position 3) at the bottom. The damaged blocks were left in cell 52-16.
- After they were removed, access channels were installed in all of the enumerated ceils. Fuel was loaded into the channels of cells 52-17 and 53-16 with incomplete graphite columns, and an additional absorber was placed in the channel of cell 52-16 with two damaged graphite blocks. The reactor continued to operate in this state without any limitations on the power.



- Initial state of cells 53-16, 52-16, and 52-17.

Comment on Contributory Impacts

Leningrad NPP

- In 1997, specialists at the Leningrad nuclear power plant developed a technological process for complete restoration of the graphite masonry of cells 52-16, 52-17, and 53-16, and they devised a system of mechanisms and hardware.
- The graphite masonry in the damaged cells was restored during the planned major overhaul.

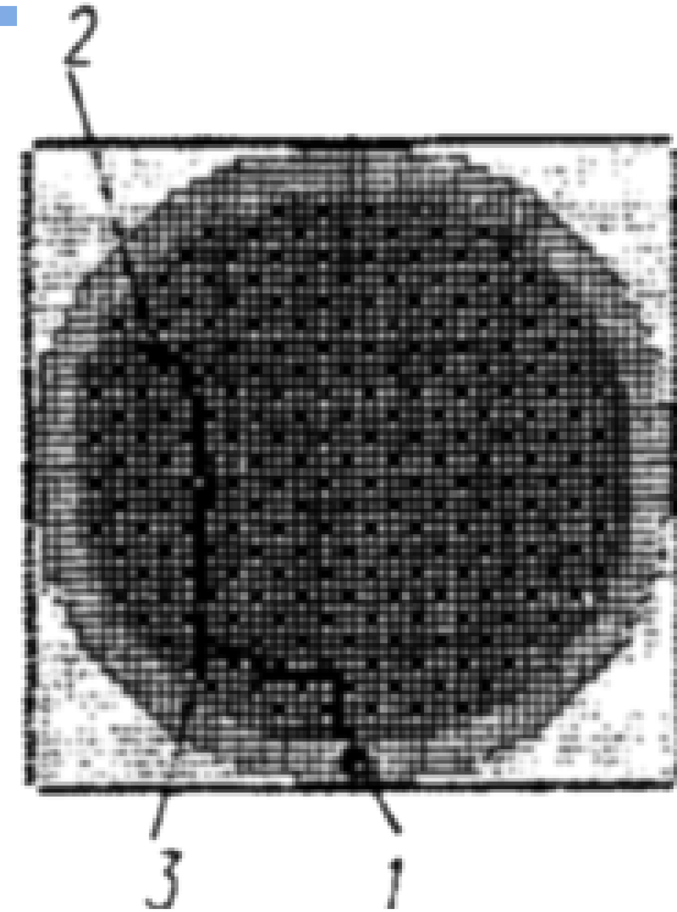
Comment on Contributory Impacts

Leningrad NPP

- A passage for the steam-gas exhaust system was installed at the top of the radiation shielding of the reactor (scheme E) in order to restore the graphite masonry during the planned maintenance work. Its location is shown in , position 1.
- The passage has a minimum diameter of 570 mm in the bottom sheet of the scheme E. Entire graphite blocks (250 • 250 x 600) and columns of blocks in an assembly (15 blocks) where transported through the passage into and out of the reactor core.
- However, the damaged cells (position 2) are located at a substantial distance fi'om the passage.
- Since during this period the access channels were replaced in the third power-generating unit and restoration of the graphite masonry requires removing the channels from the cells on which work is being performed, a decision was made to restore the masonry by moving the graphite blocks and the columns of blocks along the route devised (position 3).
- The algorithm for the solution adopted consists in the following.

Comment on Contributory Impacts

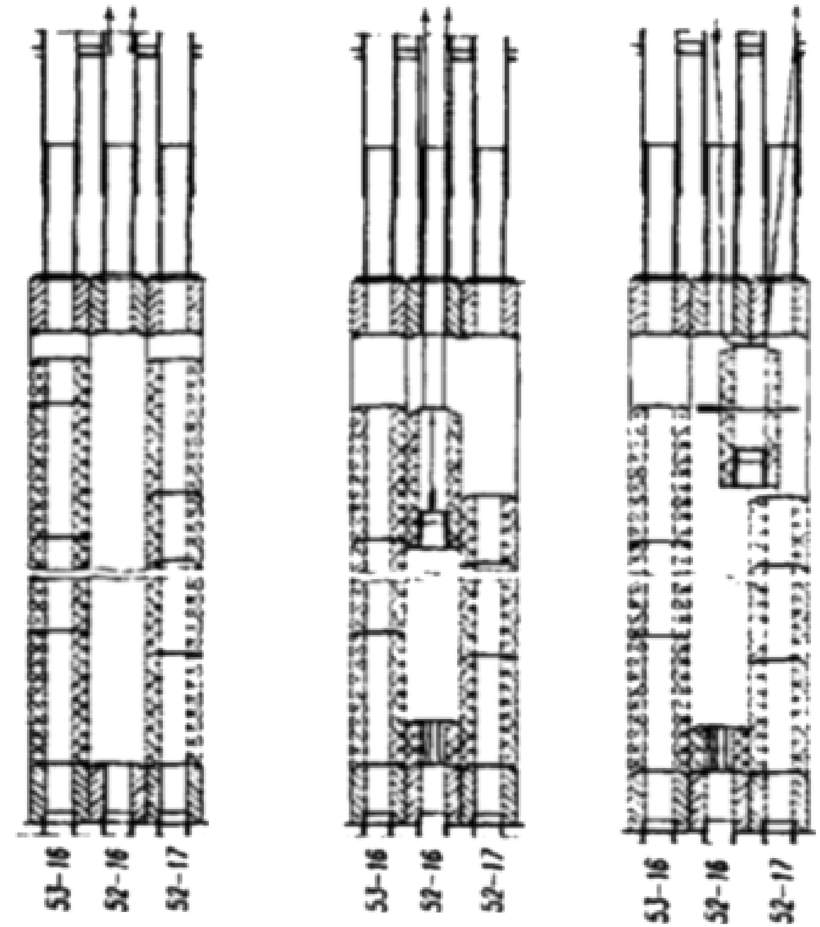
- Leningrad NPP
- The algorithm for the solution adopted consists in the following:
- The 600 mm high damaged block from cell 52-16 is put back into cell 52-17, into the cell from which it was lowered into cell 52-16.
- The 200 mm high block from cell 52-16 is put back into cell 53-16, into the cell from which it was moved into cell 52-16.
- Thus, cell 52-16 will be completely freed of graphite blocks (Fig. 3).
- Then, entire graphite columns are moved from cell to cell along the route devised (see Fig. 2, position 3), taking into account the fact that in certain ceils the access channels had already been replaced.
- The first column was moved from cell 51-16 into the empty cell 52-16. The route had an end point: the passage for the vapour-gas exhaust.



- Plateau of the reactor

Comment on Contributory Impacts

- Leningrad NPP
- Cell 52-16, freed of graphite blocks
- Lifting of a 600 mm high block
- Moving the 600 mm high block into cell 52-17.



Comment on Contributory Impacts

- Leningrad NPP
- Three types of toilet grips, differing by the height of the housing, were developed for moving the graphite blocks and entire columns.
- A special key with a built-in KTP- 155 TV camera was developed in order to lower a grip to the required marker and to control the collet grip.

Comment on Contributory Impacts

- Leningrad NPP
- While it was being moved, the 200 mm high block broke up because of cracks present in it. It had to be broken into small pieces and removed from the core through the bottom channel. Subsequently, it was restored into cell 53-16 in graphite segments.

Comment on Contributory Impacts

- Leningrad NPP
- In the course of the work, a decision was made to extract from the core columns of entire graphite blocks of cell 12-36 in order to investigate the state of the graphite.
- For this, the following operations were performed after the columns were moved from cell 13-36 into cell 14-36:
  - The graphite rods where extracted, using a special vacuum suction device, from the opening in the columns of cells 07-37 and 06-37 in the reflection zone, and the columns of blocks in cells 06-37 and 07-37 where extracted, using the collet grip, through the vapour-gas exhaust passage.
  - The column of cell 10-37 was moved into cell 07-37 and extracted.
  - Moving successively the column of blocks from cell in cell, the columns of cells 11-36 and 12-36 were also extracted from the core.
- In total of two columns where extracted from the reflection zones and four columns were extracted from the core.

Comment on Contributory Impacts

- Leningrad NPP
- Columns of new graphite blocks were placed in cells 13-36 and 12-36 by successively moving the columns from cell to cell. The columns of graphite blocks from cells 11-36, 10-36, 10-37, 07-37, and 06-37 were moved into their proper positions.
- The columns where extracted and positioned through the steam-gas exhaust passage. The column of graphite blocks from cell 12-36 was shipped to IRTM of the Russian Science Center "Kurchatov Institute" for study.

Comment on Contributory Impacts

- Leningrad NPP
- **The external gamma radiation made the main contribution to the dose load to the personnel.**
- **The gamma-ray dose rate at the work stations did not exceed the average value tbr the reactor plateau and was equal to 2 ~R/sec.**
- **The gamma-ray dose rate next to the extracted graphite column of cell 12-36 was 150-300 ~R/sec, on the average, over the entire height of the column.**
- **The maximum value 30 p.R/sec was observed on the graphite blocks 10 m from the top.**
- **The indefinite surface contamination of a graphite column with ~ and 13 emitting nuclides was -70 and 2000 particles/(cth2.min), respectively.**
- **A radionuclide analysis of smears taken from surface of a graphite column showed that 60% of the surface contamination was due to 60Co and 40% due to 134'137Cs and 144Ce.**
- **The collective equivalent dose for the entire period of work from September 2 to December 2, 1997 reached 19.88 man-rem.**
- **The maximum individual dose was -1.69 rem, and the average dose was -1. 1 rem.**
- **The allowed dose was not exceeded during the conduct of the work.**

Comment on Contributory Impacts

- Leningrad NPP
- For the first time in the history of the operation of RBMK reactors, an entire column of graphite blocks was removed from the core.
- Investigation of these blocks will provide answers for many questions concerning the behaviour of the graphite masonry during operation.
- These investigations are of great importance for the safety and reliability of power-generating units and for increasing their service life.
- The technology, developed and tested in practice, for restoration and replacement of the graphite masonry of an RBMK reactor shows that the masonry can be repaired without limiting the service life of the reactor.

Comment on Contributory Impacts

- Potential methods of removal

- Any current options being considered

- IGNALINA NPP. First Option: Dismantling of the Channels and Removal of the Graphite Columns with the Upper Shielding in Place (FDP for INPP)

- This sequence enables:
  - *the dismantling of the channels i.e. the most radioactive components of the reactor and*
  - *the removal of the graphite bricks*
  - *by implementation of already used techniques during maintenance and repair outages of RBMKs.*
  - *an important feature with respect to the operator's exposure lays into the fact that the upper shielding structure remains in place during the dismantling of those components.*

Comment on Contributory Impacts

- <u>First Option</u>
- The sequence involves the following steps:
- Removal of reactor channels. The existing technology, equipment and tools used during outages will be implemented.
- INPP records of events having led to reactor channel leakages are of high importance and will be analyzed in the very early stage of the reactor dismantling preparatory works in order to plan a cost effective dismantling activity;
- neutron and photon scanning technique will be implemented if necessary.
- Dismantling of central hall plate floor at the places of steam-gas release pipes entries into the top metal structure.
- Cutting of the steam-water pipes above the reactor in the zone of their entries (portable band saws).
- Cutting of the steam-gas release pipe works (external cutters).

Comment on Contributory Impacts

- First Option:
- **Removal of graphite stack columns (except reflector columns without channels).**
- **A procedure and technique elaborated and implemented at Leningrad NPP for the reconstruction of the graphite stack after fuel channel incident, can be used.**
- **The main idea consists of sequential shifts of the graphite columns from one location of the graphite stack to another and their final removal through the steam-gas release hatches in the top metal structure.**
- **At first, shield steel plates of neighbour cells are lifted and fixed in upper position by special equipment.**
- **A collet clutch is inserted into the central hole of a graphite column via the empty channel path by two steel ropes.**
- **The appropriate position of the clutch is controlled by TV camera.**
- **The clutch is attached to the graphite column and is subsequently lifted together with the graphite column using one rope to a a height sufficient for moving.**
- **After that the other rope is shifted to the neighbouring stack position by means of a special hook.**
- **When pulling this rope and releasing the first one, the column is moving to the neighbouring cell.**
- **The procedure starts with the removal of the column right underneath the steam-gas release hatch.**
- **By utilizing all of these hatches the procedure can be speeded up significantly.**
- **The inserted graphite rods in the lateral reflector columns are removed by a special vacuum plug.**

Comment on Contributory Impacts

- First Option:
- Complete dismantling of the central hall plate floor.
- Cutting of the remaining steam-water pipes and reactor channels paths above the upper biological shield (portable band saw, external cutters).
- Removal of the top metal structure filling (the technology used during modernization of steam-gas mixture release system can be implemented for removal of the serpentinite filling – “vacuum cleaner”).
- Dismantling of the top metal structure and removal of remaining graphite columns.

Comment on Contributory Impacts

- First Option:
- The removal of the core graphite stack without preceding dismantling of top metal structure reduces the irradiation exposure of the personnel.
- However, this technology leads to increase the duration of work. To smoothen this disadvantage, the parallel works through some hatches can be implemented.
- A second option, using remotely operated tools for the graphite handling, is proposed hereafter.

Comment on Contributory Impacts

- First Option:
- *Graphite bricks handling* (see also option 2 hereafter)
- Containers to be loaded with graphite bricks can be installed in the central hall at level 25.20 m (see loading area A.
- During the preparatory works, it shall be verified that the floor of the proposed loading area can withstand a load of about 2 t/m2.
- The loaded containers are applied with their lid, lifted up with the central hall crane and evacuated via the containers evacuation hatch (Item B).

Comment on Contributory Impacts

- IGNALINA NPP. Second Option: Dismantling of the Channels with the Upper Shielding in Place – Removal of the Graphite Columns with a Remotely Operated Graphite Blocks Handling Grab
- The first step of this second option is similar to that of the first option (dismantling of the channels with the existing tools);
- The second step consists of the removal of part of the top shielding to gain access to the graphite stack.
- This is a critical stage of the reactor dismantling. A temporary shielding ("gamma gate") may be required to provide the needed protection.
- This temporary shielding ("gamma gate") will enable the introduction of the remotely operated graphite blocks handling grabs (Figure 9.18) for the lifting and handling of the graphite blocks.
- The aim is that the grab is inserted into the bore of the graphite bricks, i.e. in this case 114mm.
- By hydraulic or pneumatic means, the balls are expended to grip the bore.
- The graphite blocks are then lifted up by the central hall crane and transported to and remotely released into the evacuation containers .
- This type of graphite blocks grabs was widely used for the construction of graphite cores in the UK reactor and for the dismantling of Windscale AGR.
- Another type of grab used at Windscale AGR involved drilling holes into the top of the graphite block, where there is no bore (i.e. solid reflector block).
- The grab then creates a screw thread in the holes by tapping, and was found to be a very reliable method of lifting the blocks, better than the vacuum grab.

Comment on Contributory Impacts

- Equipment to be kept in service:
  - *Whatever the dismantling option will be, a number of existing operation, maintenance and repair equipment will have to be kept in operation till the dismantling of the reactor or installed for that purpose:*
  - *the fuel channels cutting and removal equipment;*
  - *the channel segmentation facility (FFLA), with improved throughput rates or, preferably, a new unit;*
  - *the central hall crane and hoisting devices.*

  - *Note:*
  - **The graphite set removal facility for Fuel Channels (FC) and Control and Protection System Channels (CPS C).**
    - *The graphite set removal facility is meant for removal of graphite sleeves from fuel channels сб.12 and working channels CPS сб.14, as well as for loading of removed graphite to the container for radioactive waste. The FC and CPS C, meant for utilization, are transported to the CH from the SPH by the RM transfer car.*
    - *The utilized FC or CPS C is taken out of the storage by the CH crane with carrying capacity 50/10t, is freed from the graphite set in the graphite set removal facility, then it is installed in Facility for fragmentation of long articles for fragmentation.*

Comment on Contributory Impacts

- Preparatory Works
  - *It can be expected that the dismantling of INPP reactors will be the first large size industrial RBMK to be dismantled in the world.*
  - *The Russian and Ukrainian RBMKs will be dismantled in accordance with the different dismantling option, i.e. 50 years after the RFS.*
- The preparatory works to be conducted during the next 10-12 years will include:
  - *the development (design, manufacturing testing) of the remotely operated tools to carry out the dismantling operation sequence;*
  - *the improvement (or the replacement) of the channel segmentation facility;*
  - *the construction of a single mock-up of the reactor or mock-up of specific parts of the reactor. This (or these) mock-up(s) will enable to optimize the dismantling sequences and tools, as well as the training of the operators;*
  - *the selection of the tools and locations to carry out the final dismantling of the reactor pre-cut components;*
  - *the transportation ways of the dismantled equipment between the reactor hall and the final cut places, including the need for new openings and handling means (cranes, hoisting, devices).*

Comment on Contributory Impacts

Graphite	tons	m³
Operational waste	55	46
Decommissioning waste: graphite bricks and sleeves for 2 units	3788	2945
Total	**3843**	**2991**

- On the basis of container internal dimensions of 2.8 m x 1.45 m and a height of 1.01 m, it is assumed that an average of 2.625 m^3 of graphite bricks could be placed inside one container (estimation basis: bricks of 0.25 m x 0.25 m x 0.6m only are considered, 50 being placed vertically on the bottom of the container and 20 being placed horizontally on top of the first layer).
- About 1140 containers would be required to pack the graphite waste. The external volume of these containers to be interim stored on the INPP site would represent some 7054m^3.

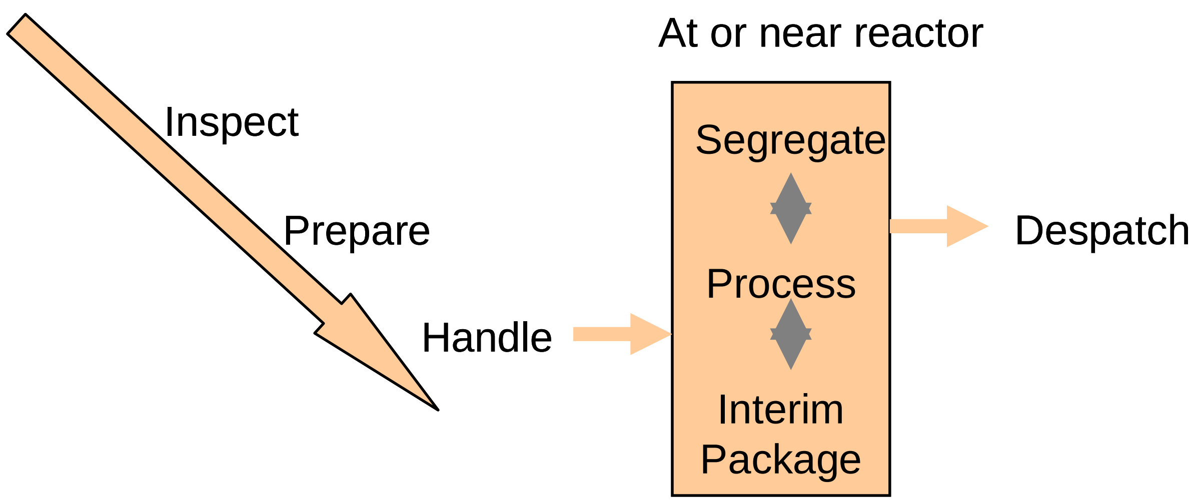
Comment on Contributory Impacts

- Other materials e.g. thermocouples, securing wires that might be encountered
  - Thermocouples will be removed before removal of the graphite
- Implications of reactor design on graphite retrieval
  - Especially any known complications not immediately obvious from the drawings: such information is not available

Decision Support Data

- Any other issues you may wish to raise that will facilitate of support the arrival at an i-graphite retrieval and segregation solution
- Any regulator comment on i-graphite management
  - Special regulations are not available.
- Any relevant cost data available e.g. budgets
  - Not available
- Any additional information arising from the information needs identified in Table 3 of the Data Pack
  - Note: no comments received to date

Flow Diagram



- Comment on this diagram with respect to
  - Magnox or
  - RBMK or
  - UNGG
- Update diagram is necessary
- Answer question: What impact has the information on the previous slides likely to have on this generic process?

Needs from Workshop

- With respect to the current state of knowledge on (Magnox, RBMK or UNGG)
  - Comment on current state of i-graphite retrieval (mode of dismantling) with respect to your reactor design (where are you at in the planning, safety case, implementation process):
  - Two preliminary options of reactor dismantling in FDP for Unit 1
  - Tendering for feasibility study on dismantling of the reactor
- What tools (in context of CARBOWASTE output) are still needed?
  - Comment on design/planning, modelling and implementation needs
  - Who needs the tools? Operators, regulators, contractors others?
  - Try and differentiate between the needs of these groups

APPENDIX 4 Possible Options for Ignalina NPP RBMK-1500 Reactors Dismantling

It comprises a report prepared as an action following the Work Package 1 / 2 workshop held in Manchester in February 2009.

The document was prepared by Poskas & Poskas of the Lithuanian Energy Institute.

It is presented un edited.

P. Poskas, R. Poskas
Lithuanian Energy Institute

1. POSSIBLE OPTIONS FOR IGNALINA NPP RBMK-1500 REACTORS DISMANTLING

1.1. EXPERIENCE ON RESTORATION OF THE GRAPHITE CORE AT LENINGRAD NPP UNIT 3 [1].

The graphite masonry of an RBMK-1000 reactor is a basic stucture that determines the service life of the reactor. However, the construction of the reactor unit does not permit repair and maintenance work to be performed on the unit.

Specialists at the Leningrad nuclear power plant were tasked with solving the following problem:

- to develop a technology that would make it possible to extract from the reactor an entire graphite column for shipment to the Russian Science Center "Kurchatov Institute" and
- to restore the graphite core of the reactor in the third power-generating unit, in which cell 52-16 operated without a graphite colunm „after the liquidation of the consequences of a loss of seal in the access channel“ in March 1992.

In 1992, during the liquidation of the consequences of the loss of seal in the access channel, the top steel plates of cells 52-16, 52-17, and 53-16 of assembly 07 were lifted into the extreme top position and clamped to the pipes in the top sections. All 14 graphite blocks were removed from cell 52-16 by grinding. In order to remove the fragments of graphite and fuel assemblies from the gaps, the adjoining columns were alternately moved, block by block, into the empty cell 52-16. Next, these columns were put back into their intended position. However, one 200 mm high block and one 600 mm high block where damaged when the blocks were moved. The damaged blocks were left in cell 52-16 (Fig. 1).

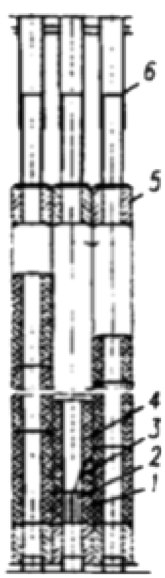


Fig. 1. State of cells 53-16, 52-16 and 52-17 in 1992 after liquidation of the consequences of the loss of seal in the acces channel.

After that, fuel channels were installed in all of the enumerated cells. Fuel was loaded into the channels of cells 52-17 and 53-16 with incomplete graphite columns, and an additional absorber was placed in the channel of cell 52-16 with two damaged graphite blocks. The reactor continued to operate in this state without any limitations on the power.

In 1997, specialists at the Leningrad nuclear power plant developed a technological process for complete restoration of the graphite core. The graphite masonry in the damaged cells was restored during the planned major overhaul.

A passage for the steam-gas exhaust system was installed at the top of the radiation shielding of the reactor (scheme E) in order to restore the graphite masonry during the planned maintenance work. Its location is shown in Fig. 2, position 1. The passage has a minimum diameter of 570 mm in the bottom sheet of the scheme E. Entire graphite blocks (250 • 250 x 600) and columns of blocks in an assembly (15 blocks) where transported through the passage into and out of the reactor core.

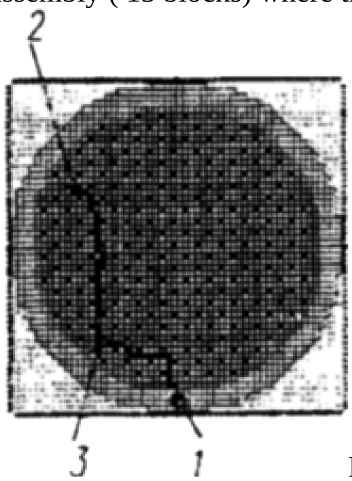
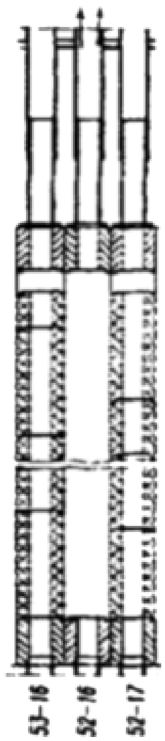
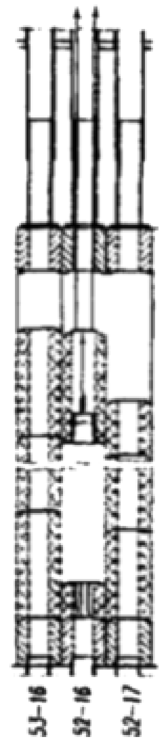


Fig. 2. Plateau of the reactor.

However, the damaged cells (position 2) are located at a substantial distance from the passage. Since during this period the access channels were replaced in the third power-generating unit and restoration of the graphite masonry requries removing the channels from the cells on which work is being performed, a decision was made to restore the masonry by moving the graphite blocks and the columns of blocks along the route devised (position 3).

Three types of toilet grips, differing by the height of the housing, were developed for moving the graphite blocks and entire columns. The grip for the 200 and 300 mm high graphite blocks was 200 mm high, the grip tbr the 500 and 600 mm high blocks was 500 mm high, and the grip for the columns of graphite blocks was 8000 mm high. Two 6 mm thick steel cables were threaded into the top part of a grip. The length of the cables was sufficient to lower the grips to the bottom graphite block. A special key with a built-in KTP- 155 TV camera was developed in order to lower a grip to the required marker and to control the collet grip. The free ends of the steel cables where wound on winch drums placed next to the cell. The schemes for the transfer of the graphite blocks is shown in Fig. 3-5.





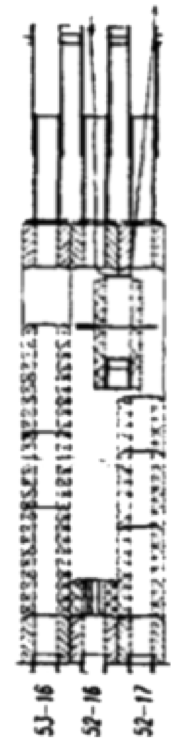


Fig. 3. Cell 52-16, freed of graphite blocks	Fig. 4. Lifting of a 600 mm high graphite block	Fig. 5. Moving the 600 mm high block into cell 52-17

While it was being moved, the 200 mm high block broke up because of cracks present in it. It had to be broken into small pieces and removed from the core through the bottom channel. Subsequently, it was restored into cell 53-16 in graphite segments.

It should be noted that the top steel blocks (see Fig. 1, position 5) of cells 52-16, 52-17, and 53-l 6 were clamped to the pipe in the top channel by welding (position 6). A setup for raising and fixing the assembly 07 in the top position without welding was used to move the columns of other cells.

After the columns of graphite blocks were put back into cells 52-16, 52-17, and 53-16, the weld seams clamping the assembly 07 to the pipe in the top section were cut off and the assembly was lowered into its proper postion.

In the course of the work, a decision was made to extract from the core columns of entire graphite blocks of cell 12-36 in order to investigate the state of the graphite. For this, the following operations were performed after the columns were moved from cell 13-36 into cell 14-36. The graphite rods where extracted, using a special vacuum suction device, from the opening in the columns of cells 07-37 and 06-37 in the reflection zone, and the columns of blocks in cells 06-37 and 07-37 where extracted, using the collet grip, through the vapor-gas exhaust pasage. The column of cell 10-37 was moved into cell 07-37 and extracted. Moving successively
the column of blocks from cell in cell, the columns of cells 11-36 and 12-36 were also extracted from the core. A total of two columns where extracted from the reflection zones and four columns were extracted from the core. Columns of new graphite blocks were placed in empty cells.

Organizational-technical measures were taken to ensure radiation safety of the work. The technology was worked out on a full-scale stand, where the weaknesses of the technological process and the specialized hardware were determined and eliminated. The personnel were trained and became skilled in this work.

The work on the reactor unit was performed according to an operations document, using instantaneous KID-6 dosimeters (in addition to the main TLD dosimeters) and RADOS direct-indication dosimeters with continuous monitoring of the "/-ray dose rate in the central room using the AKRB-06 "Gorbach" automatic system. During the operation, the dosimetrist on duty constantly monitored the parameters of the radiation environment directly at the work stations. The KID-6 and RADOS dosimeters made it possible to monitor rapidly the "daily dose load to the personnel. The dose by operation and the daily dose were monitored using a computer system based on a data bank of the individual dosimetric monitoring.

To prevent radioactive contaminants from escaping from the central room, sanitary barriers with washing of footwear and hands and with individual means of protection were organized. The floor in the central room around the reactor plateau was covered with plasticized rubber. The specialized equipment was decontaminated at the end of each shift by washing with water and a "Radez" preparation. The washing solutions were discarded in a special channel in the central room. The specialized equipment was stored in polyethylene sleeves in a special segregated location in the central room.

So, for the first time in the history of the operation of RBMK reactors, an entire column of graphite blocks was removed from the core. Investigation of these blocks will provide answers for many questions concerning the behavior of the graphite core during operation. These investigations are of great importance for the safety and reliability of power-generating units and for increasing their service life. The technology, developed and tested in practice, for restoration and replacement of the graphite columns of an RBMK reactor showed that the graphite core can be repaired without limiting the service life of the reactor. Experience gained is also very important for planning of the dismantling of the RBMK reactors.

1.2. OPTIONS FOR IGNALINA NPP RBMK-1500 REACTORS DISMANTLING [2].

In the Final Decommissioning Plan (FDP) for Ignalina NPP Unit 1 and Unit 2 only a general view of the sequence of activities, dismantling methods and tools is presented. Immediate dismantling strategy for RBMK-1500 reactors at Ignalina NPP is selected, so due to the high dose rates, remotely operated tools and robots will be needed for the dismantling of the reactor components. The contamination of the graphite stack with fuel particles can significantly increase the graphite stack dose rate (up to 10 Sv/h after 100 years decay). But, such type of incidents is not expected to be a serious concern at INPP. With the exception of the already existing fuel channels cutting tools, this equipment will have to be especially designed or adapted, taking into account the characteristics of the reactor components. General view of reactor RBMK-1500 cross-section, top metal structure and plan of reactor hall are given at the Fig. 7-9.

Prior to starting the dismantling, the reactor should be completely defuelled, the control rods and in-core instrumentation be removed and the lateral biological shield should be drained down.

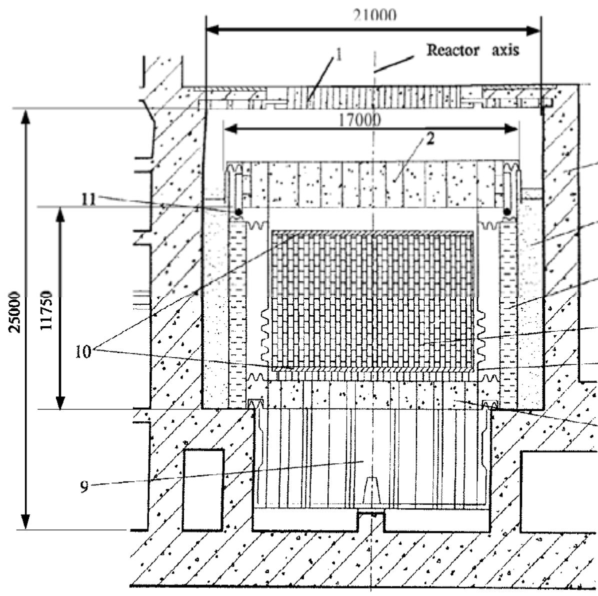


Fig. 7. General view of reactor RBMK-1500 cross-section.
1 – plate floor, 2 – top metal structure with serpentinite filling, 3 – concrete vault, 4 – mounting space with sand filling, 5 – lateral biological shield (annular water tank), 6 – graphite stack, 7 – reactor case, 8 – bottom metal structure with filling, 9 – reactor support structure, 10 – support (bottom) and shield (top) steel plates, 11 – roller supports

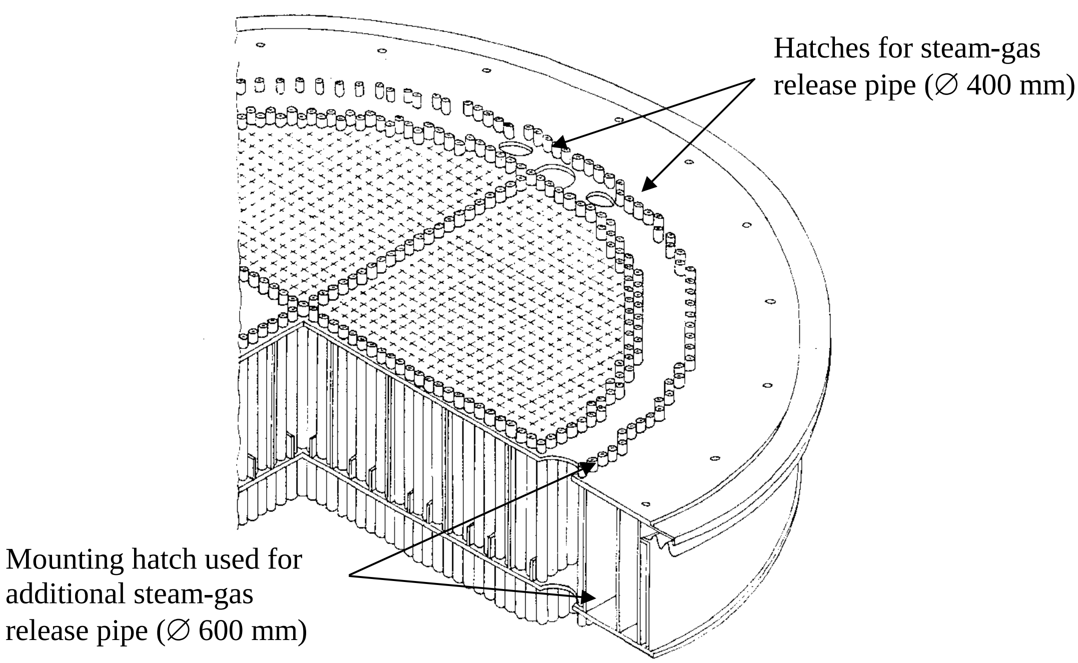


Fig. 8. Top metal structure (structure E).

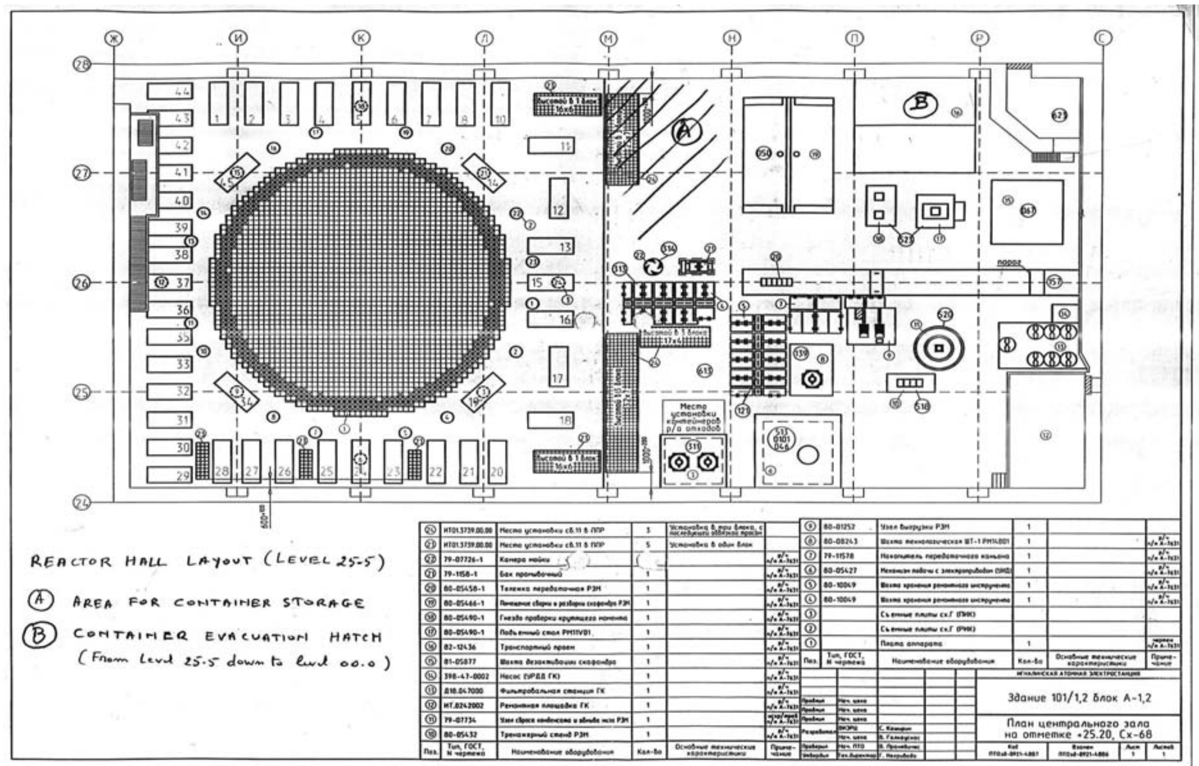


Fig. 9. Plan of Reactor Hall. A – containers loading area, B - containers evacuation hatch.

1.2.1. First Option: Dismantling of the Channels and Removal of the Graphite Columns with the Upper Shielding in Place

This sequence enables the dismantling of the channels i.e. the most radioactive components of the reactor and the removal of the graphite bricks by implementation of already used techniques during maintenance and repair outages of RBMKs. As hereabove indicated, an important feature with respect to the operator's exposure lays into the fact that the upper shielding structure remains in place during the dismantling of those components.

The sequence involves the following steps:

- Removal of reactor channels. The existing technology, equipment and tools used during outages will be implemented. The final segmentation of channels will be a critical operation due to low output of existing facility for segmentation of extended components (FFLA). To satisfy the decommissioning time constrains, the installation should be modernized or a new one should be constructed.

  In order to avoid the generation of "bottlenecks" and the interruption of the channels cutting process, the removed channels can be interim stored in defueled pools, awaiting upon further segmentation in the FFLA.

- In order to obtain accurate data about the possible contamination of the graphite stack with fuel particles and fission products (localization and magnitude of the contamination):

- INPP records of events having led to reactor channel leakages are of high importance and will be analyzed in the very early stage of the reactor dismantling preparatory works in order to plan a cost effective dismantling activity;
- neutron and photon scanning technique will be implemented if necessary.

- Dismantling of central hall plate floor at the places of steam-gas release pipes entries into the top metal structure (Figure 9.10).
- Cutting of the steam-water pipes above the reactor in the zone of their entries (portable band saws).
- Cutting of the steam-gas release pipe works (external cutters).
- Removal of graphite stack columns (except reflector columns without channels). A procedure and technique elaborated and implemented at Leningrad NPP for the reconstruction of the graphite stack after fuel channel incident, can be used. The main idea consists of sequential shifts of the graphite columns from one location of the graphite stack to another and their final removal through the steam-gas release hatches in the top metal structure. At first, shield steel plates of neighbour cells are lifted and fixed in upper position by special equipment. A collet clutch is inserted into the central hole of a graphite column via the empty channel path by two steel ropes. The appropriate position of the clutch is controlled by TV camera. The clutch is attached to the graphite column and is subsequently lifted together with the graphite column using one rope to a a height sufficient for moving. After that the other rope is shifted to the neighbouring stack position by means of a special hook. When pulling this rope and releasing the first one, the column is moving to the neighbouring cell. The procedure starts with the removal of the column right underneath the steam-gas release hatch. By utilizing all of these hatches the procedure can be speeded up significantly. The inserted graphite rods in the lateral reflector columns are removed by a special vacuum plug. Once finished all core components exhibiting the highest dose rate have been removed.
- Complete dismantling of the central hall plate floor.
- Cutting of the remaining steam-water pipes and reactor channels paths above the upper biological shield (portable band saw, external cutters). Removal of the top metal structure filling (the technology used during modernization of steam-gas mixture release system can be implemented for removal of the serpentinite filling – “vacuum cleaner”).
- Dismantling of the top metal structure and removal of remaining graphite columns.

The removal of the core graphite stack without preceding dismantling of top metal structure reduces the irradiation exposure of the personnel. However, this technology leads to increase the duration of work. To smoothen this disadvantage, the parallel works through some hatches can be implemented.

1.2.2. Second Option: Removal of the Graphite Columns with a Remotely Operated Graphite Blocks Handling Grab

- The first step of this second option is similar to that of the first option (dismantling of the channels with the existing tools);

- The second step consists of the removal of part of the top shielding to gain access to the graphite stack. This is a critical stage of the reactor dismantling. A temporary shielding (“gamma gate”) may be required to provide the needed protection. This temporary shielding (“gamma gate”) will enable the introduction of the remotely operated graphite blocks handling graps for the lifting and handling of the graphite blocks. The aim is that the grab is inserted into the bore of the graphite bricks, i.e. in this case 114mm. By hydraulic or pneumatic means, the balls are expended to grip the bore. The graphite blocks are then lifted up by the central hall crane and transported to and remotely loaded into the evacuation containers. This type of graphite blocks grabs was widely used for the construction of graphite cores in the UK reactor and for the dismantling of Windscale AGR. Another type of grab used at Windscale AGR involved drilling holes into the top of the graphite block, where there is no bore (i.e. solid reflector block). The grab then creates a screw thread in the holes by tapping, and was found to be a very reliable method of lifting the blocks, better than the vacuum grab.

1.2.3. Graphite handling

Containers to be loaded with graphite bricks can be installed in the central hall at level 25.20 m (see loading area A – Fig. 9). During the preparatory works, it shall be verified that the floor of the proposed loading area can withstand a load of about 2 t/m^2. The loaded containers are applied with their lid, lifted up with the central hall crane and evacuated via the containers evacuation hatch (Fig. 9 – Item B) down to the transport corridor (level 0.00) for further transportation to the final characterisation unit and to the interim storage building.

1.2.4. Equipment to be kept in service

Whatever the dismantling option will be, a number of existing operations, maintenance and repair equipment will have to be kept in operation till the dismantling of the reactor or installed for that purpose:

- the fuel channels cutting and removal equipment;
- the channel segmentation facility (FFLA), with improved throughput rates or, preferably, a new unit;
- the central hall crane and hoisting devices.

1.2.5. Preparatory Works

It can be expected that the dismantling of INPP reactors will be the first large size industrial RBMK to be dismantled in the world. The Russian and Ukrainian RBMKs will be dismantled later due to the deferred dismantling strategy selected.

The preparatory works to be conducted during the next 10-12 years will include:
the development (design, manufacturing testing) of the remotely operated tools to carry out the dismantling operation sequence;
- the improvement (or the replacement) of the channel segmentation facility;
- the construction of a single mock-up of the reactor or mock-up of specific parts of the reactor. This (or these) mock-up(s) will enable to optimize the dismantling sequences and tools, as well as the training of the operators;
- the selection of the tools and locations to carry out the final dismantling of the reactor pre-cut components;
- the transportation ways of the dismantled equipment between the reactor hall and the final cut places, including the need for new openings and handling means (cranes, hoisting, devices).

References

1. V. I. Lebedev e. a. Restoration of the reactor graphite core in the third power-generating unit of the Leningrad nuclear power plant. Atomnaya energiya, vol. 87, No. 5, pp. 356-359 (in Russian).

2. Final Decommissioning Plan for INPP Unit 1 and Unit 2. 2004, Issue 06.

APPENDIX 5 Vandellos Silos Retrieval Project Report

CARBOWASTE

Technical Report
WP2 DATA PACK

VANDELLOS 1 NPP SILOS. RETRIEVAL, SORTING AND PRECONDITIONING OF I-GRAPHITE AND OTHERS RADIOACTIVE WASTES.

Author: Francisco Madrid (ENRESA)

It comprises a report prepared for Work Package 2 as an action following the Work Package 1 / 2 workshop held in Manchester in February 2009.

The document has been presented un-edited but a review of its key contents is included in section 5.3.11 of the main report body. This example is considered a bench mark for the retrieval of unconditioned wastes from silos or vaults.

CARBOWASTE

Technical Report

WP2 DATA PACK

VANDELLOS 1 NPP SILOS. RETRIEVAL, SORTING AND PRECONDITIONING OF I-GRAPHITE AND OTHERS RADIOACTIVE WASTES.

Author: Francisco Madrid (ENRESA)

CONTENTS

1. INTRODUCTION

One issue, common to many nuclear facilities, is that the technical solution selected for the interim storage of the radioactive waste, generated during the process, consisted of the direct dropping of the wastes inside concrete silos, designed to provide suitable waste confinement for a limited period of time.

Once finished the operational period, it is required to proceed with the waste retrieval and final conditioning in a form consistent with the radiological characteristics of them.

The successful retrieval and preconditioning of the radioactive wastes loaded inside the Vandellós I NPP Silos, provides information about how to deal with the issue whenever the scenario conditions were analogous.

This report provides specific issues related to Vandellós I NPP Silos project mainly with regard to:

- Silos Geometry
- Radioactive Wastes Characteristics
- Retrieval equipment characteristics
- Transfer equipment characteristics
- Conditioning equipment characteristics
- Waste packages produced
- Auxiliary systems issues (HVAC and fire protection system)
- Project programme
- Operating experiences

The present report is organized in such a way that the specific issues are provided in the frame of a summary description of the project activities and it contains:

- Description of the silos initial conditions
- Conceptual design

  - Retrieval and transfer
  - Sorting and conditioning
  - Auxiliary systems
- Programme of the project
- Operating experiences

2. GENERAL DESCRIPTION

2.1 Silos Description

The Vandellós I NPP silos are located at one of the edges of the NPP close to the Lleira gully (see fig. 2.1.1.).

The three silos are semi-buried reinforced concrete structures and the storage capacity of each one of them is about 1.500 m3.

The silos have been built sequentially, first silo 1 (1971) and last silo 3 (1986), and the differences among them show improvements due to operating experiences.

The handling of the transfer equipment used during the operation of the NPP was carried out by means of an overhead frame crane, with a capacity of 40 tons that rolled, all along the three silos, on rails arranged on the cover slabs and over the silos lateral walls. This crane, once properly modify and updated, has been used for the handling of the retrieval and transfer equipment of the project.

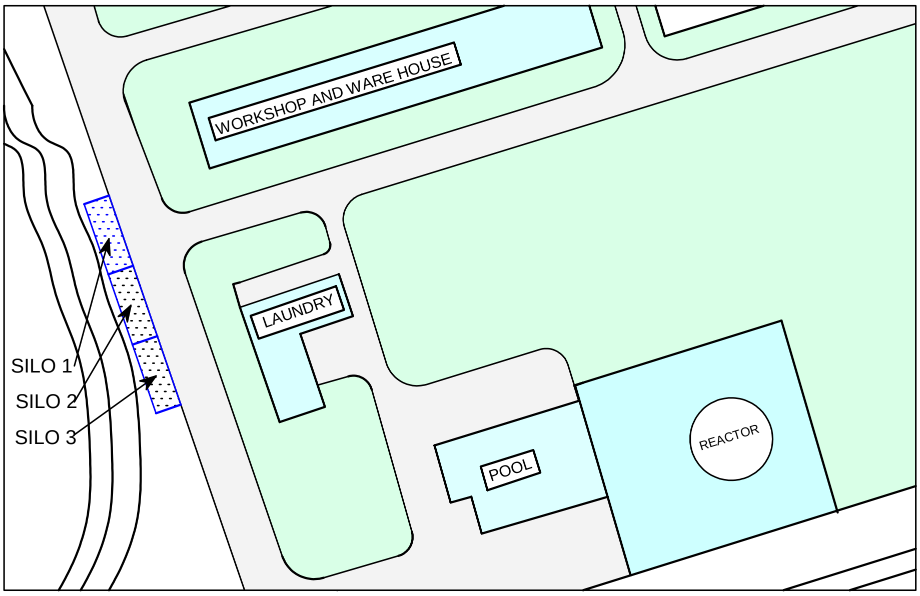


Figure 2.1.1. - LOCATION OF SILOS 1, 2 AND 3

Close enough of the silos location; the main auxiliary systems of the facility were available such as fire protection water ring, electricity, sewerage etc.

Silo 1 has the following specific geometric characteristics (see fig. 2.2.2):

- Length: 25.50 m
- Width: 8.70 m
- Height: 10.25 m

The cover slab has twelve (12) openings, used to drop the radioactive wastes into the silo, and covered by means of individuals concrete plugs.

Six (6) ventilation pipes allow the eventual extraction and ventilation of the silos internal air and a drainage well, arranged at the base concrete slab, allow the pumping of the liquid effluents up to the effluents treatment plant of the facility

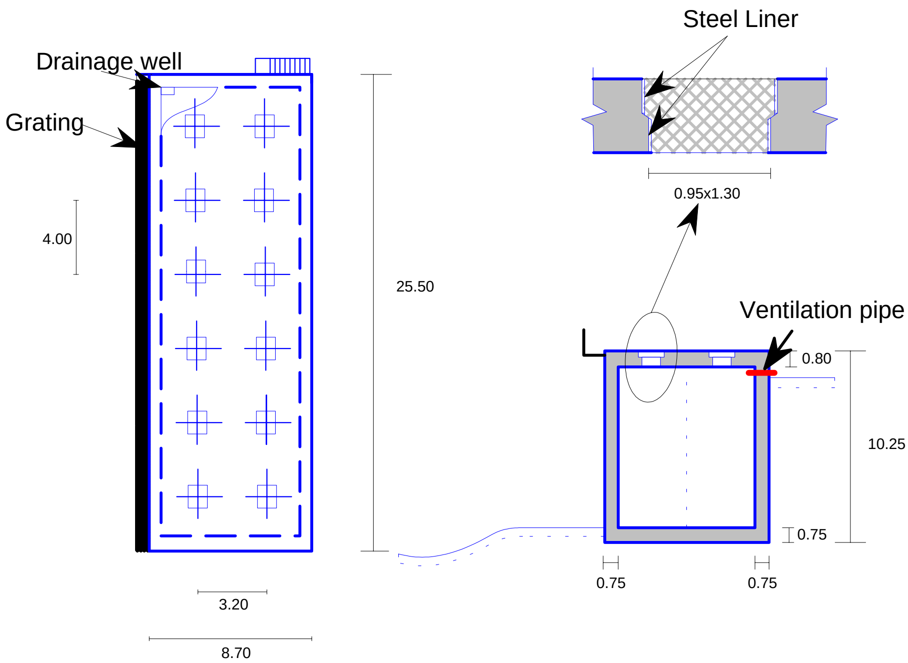


Figure 2.1.2. - SILO 1 DETAIL

During the first steps of the NPP operation, Silo 1 was the only one available and was used to store all type of wastes generated in the NPP (Graphite, Absorbents and Diverse wastes such as bags, drums with clothes, plastics, filters and others). Once silo 2 was available only diverse waste were dropped in Silo 1.

The main modifications introduced in silos 2 and 3 with regard to silo 1 were the following (see fig. 2.2.3):

- Additional shielding of the cover slab and accessible parts of the lateral walls by means of an increase of their thickness of 25 cm.
- Less unloading openings (Eight) with larger dimensions.
- Improved ventilation pipes (4 pipes for each silo)

On the bottom of silo 3, and covering 1 m of the lateral walls, a waterproof coating limited the liquid infiltration through the concrete structure. Additionally, the liquid extraction system was improved.

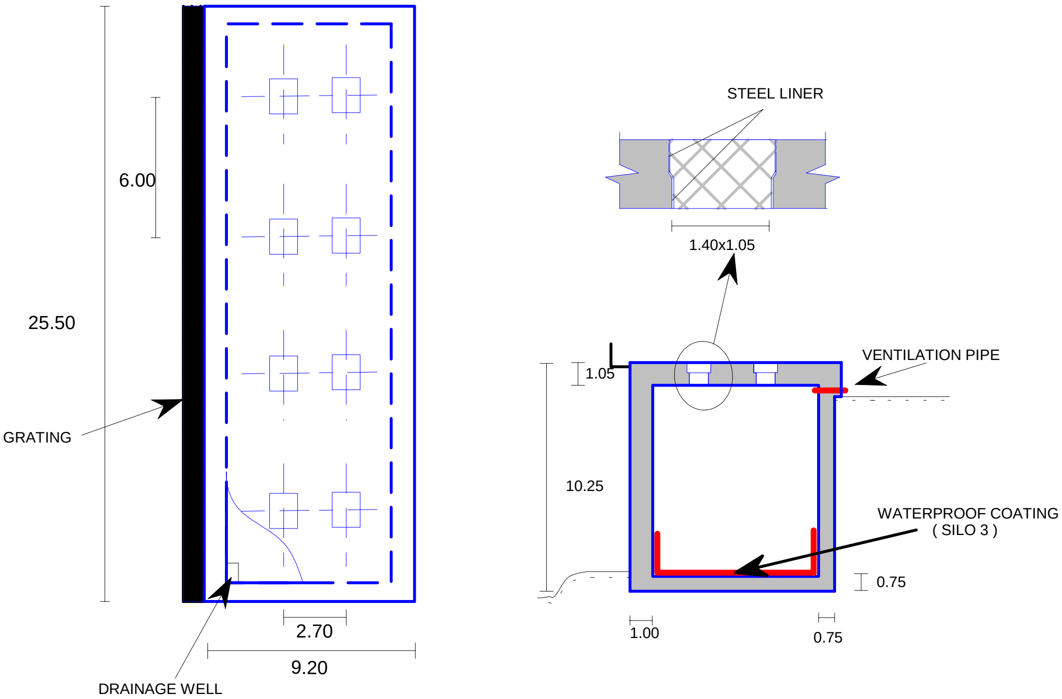


Figure 2.1.3. - SILOS 2 AND 3 DETAILS

2.2 Radioactive Wastes Description

The radioactive wastes contained in the silos are the ones typically produced during the operation of a NPP similar type as Vandellós I. Namely:

- Graphite Sleeves
- Graphite Cylinders (Rondins)
- Absorbents: Neutron flux absorbents, 'absorbents' hereinafter in this document are dummy fuel elements that are placed in reactor core to shape the neutron flux profile. They are identical to the fuel elements excepting that they have a stainless steel core instead of a uranium core.
- Seat support wires (stainless steel wires)
- Diverse wastes (bags and drums with clothes, plastics, filters and others)

Additionally two spent fuel elements, with its graphite sleeves, had been accidentally dropped inside silo 1.

The inventory of wastes was:

TYPE	WASTE INVENTORY		
	SILO 1	SILO 2	SILO 3
Graphite Sleeves	36.123	106.666	43.628
Graphite Cylinders	4.834	54	-
Absorbents	210	282	-
Seat support wires	36.614	108.165	43.628
Diverse wastes (bags and drums)	929	-	-
spent fuel elements	2		

TABLE 2.2.1 WASTE INVENTORY

Taking into account the waste inventory and the activity calculated for each of the waste types, the source terms considered in the project for each of the silos, and consequently for the various streams of wastes, are shown in the following tables.

Nuclide	Support wires	Graphite Sleeves	Absorbents	Rondins	Drums/Bag		Spent Fuel	Total
	Bq/u	**Bq/KG**	**Bq/u**	**Bq/u**	**Type 1**	**Type 2 y**	**Bq/u**	**Bq**
Co–60	2.065E+09	1.557E+06	6.204E+10	8.327E+06	1.972E+0	1.792E+04		8.359 E+1
Ni–63	2.362E+09							8.175 E+1
Ni–59	2.133E+07							7.384 E+1
Nb–94	7.588E+05							2.627 E+1
C–14		1.600E+07		1.532E+07				3.195 E+1
Ca–41		1.195E+04		5.358E+04				2.590 E+0
Cs–137		4.431E+04		1.663E+08			1.277E+1	3.367 E+1
Sr–90		1.127E+07		1.169E+08			1.274E+1	5.311 E+1
Pu–241							3.631E+1	7.263 E+1
Kr–85							2.619E+1	5.237 E+1
Pu–239							2.560E+1	5.120 E+1
Pu–240							1.566E+1	3.132 E+1
Pu–238							2.480E+0	4.960 E+0
Am–241							6.418E+0	1.284 E+0
Am–242M							3.789E+0	7.578 E+0
U–238							1.110E+0	2.220 E+0
Cm–244							2.998E+0	5.995 E+0
Cm–243							1.392E+0	2.784 E+0
U–236							5.920E+0	1.184 E+0
U–235							4.440E+0	8.880 E+0
Am–243							2.958E+0	5.917 E+0
Pu–242							2.590E+0	5.180 E+0
Np–237							2.220E+0	4.440 E+0
I–129							9.657E+0	1.931 E+0
Total	**4.449 E+09**	**2.888 E+07**	**6.204 E+10**	**3.069 E+08**	**1.972 E+0**	**1.792 E+1**	**3.221E+1**	**1.811 E+1**
Total Units	34614	36123	210	4834	909	20	2	
Weight (kg)	3.07 E+02	1.95 E+05	2.68 E+03	4.83 E+04	4.55 E+0	1.00 E+0	3.41 E+0	
Total act,.	1.540 E+14	5.633 E+12	1.303 E+13	1.483 E+12	1.792 E+1	3.584 E+1	6.443 E+1	

Table 2.2.2. SILO 1 SOURCE TERM

Nuclide	Support Wires (Bq/u)	Graphite Sleeves (Bq/kg)	Absorbents (Bq/u)	Rondins (Bq/u)	Total Activity Bq
Cr-51	3.858E-10				4.173E-05
Co-60	9.490E+09	3.023E+06	1.473E+11	6.332E+07	1.070E+15
Ni-63	4.424E+09				4.785E+14
Ni-59	3.649E+07				3.947E+12
S-35	1.212E-01				1.311E+04
Nb-94	1.402E+05				1.517E+10
C-14		1.600E+07		8.697E+07	9.220E+12
Ca-41		2.248E+04		2.741E+05	1.296E+10
Cs-137		5.022E+04		1.896E+08	3.916E+10
Sr-90		1.372E+07		1.348E+08	7.910E+12
TOTAL	**1.395E+10**	**3.281E+07**	**1.473E+11**	**4.750E+08**	1.569E+15

Total Units	108165	106666	282	54
Weight (kg)	9.59E+02	5.76E+05	3.60E+03	5.40E+02
Total Act.(Bq)	1.509E+15	1.890E+13	4.153E+13	2.565E+10

TABLE 2.2.3 SILO 2 SOURCE TERM

Nuclide	Support Wires Bq/u	Graphite Sleeves Bq/kg	Total Activity Bq
Co-60	1.993E+10	3.756E+06	8.706E+14
Ni-63	4.646E+09		2.027E+14
Ni-59	3.702E+07		1.615E+12
C-14		1.600E+07	3.769E+12
Cs-137		3.840E+05	9.048E+10
Sr-90		1.594E+07	3.755E+12
S-35		1.106E+01	2.606E+06
Total	**2.462E+10**	**3.608E+07**	**1.083E+15**

Total Units	43628	43628
Weight (kg)	3.870E+02	2.356E+05
Total Act.(Bq)	1.074E+15	8.500E+12

TABLE 2.2.4 SILO 3 SOURCE TERM

3 WASTE RETRIEVAL AND CONDITIONING CONCEPTUAL DESIGN

3.1 General Flow Diagram

The design was carried out to be consistent with the following main restrictions:

Existing restrictions:

- Silos Configuration.
  - Similar configuration and waste accessibility for each of the silos
- Waste Characteristics
  - Diverse wastes in silo 1 and graphite in silo 1, silo 2 and silo 3.
  - Intricate link between graphite and support wires in the graphite sleeves.
  - Two fuel elements were inadvertently dropped inside silo 1 and the design had to be compatible with this scenario (the description of the especial measures taken to detect, retrieve, and dispatch the fuel elements are not included in the report).
  - Excepting the above mentioned two fuel elements in Silos, there were not uranium in Silos 2 and 3.
  - No significant presence of water, coming from rainfall infiltration, inside the silos. There was not significant amount of sludges.
- Fullness status and wastes configuration inside of each silo
  - No especial risk of waste collapse.
  - The wastes in silo 2 did not allow the use of the petal grab at the beginning of the retrieval operation.

Packages to be produced

The types of packages to be produced, as a result of the conditioning process, were decided by ENRESA (The Radioactive Waste Management Spanish Agency) taking into account:

- The wastes physical and radiological characteristics

- The waste conditioned process and the packages were designed to allow, without major restrictions, further conditioning treatments of the wastes. Consequently the uses of technologies such as grouting were not considered as suitable conditioning process.

Additionally the design had to use existing, reliable and proven technologies and had to comply with the radiological requirements applicable to nuclear facilities.

Figure 3.1.1 shows the main activities of the process, as follows:

- Visual selection (CCTV) and retrieval
- Transfer from silos to the conditioning building
- Conditioning
- Transfer for interim storage

Sections 3.1 and 3.2 deal with a description of the most significant items of the main process while section 3.3 describes the generic of the auxiliary systems.

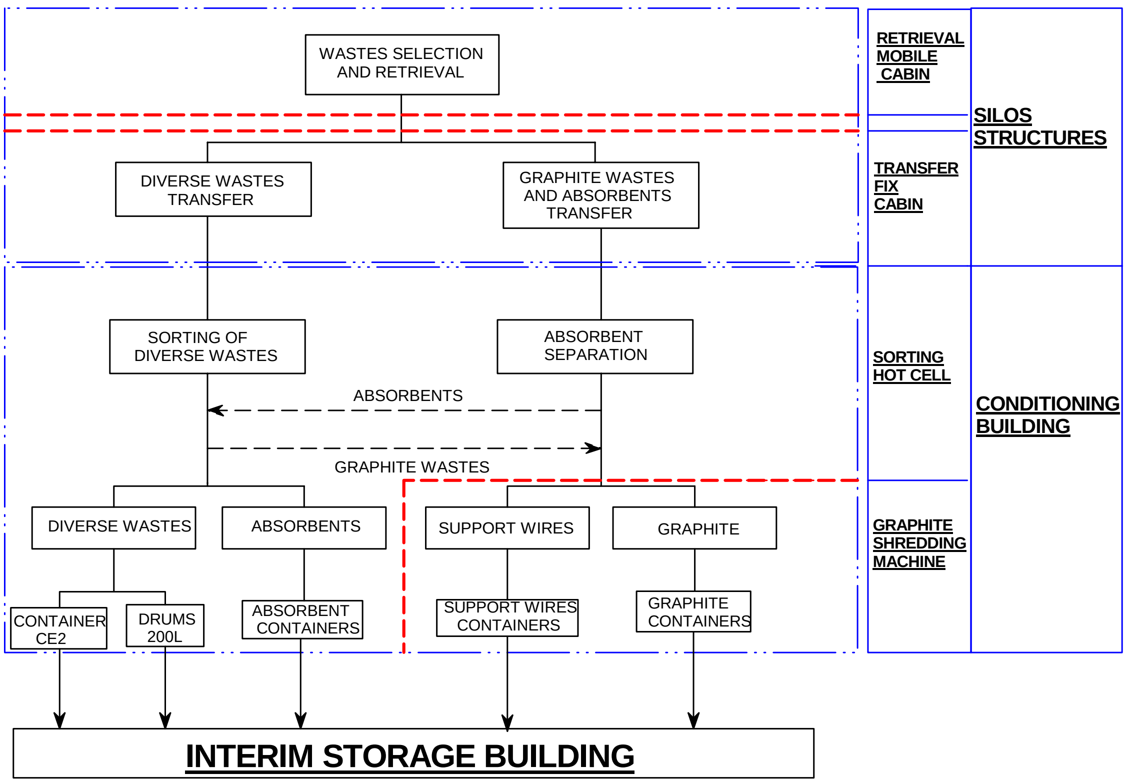


FIGURE 3.1.1. PROCESS FLOW DIAGRAM

3.2 *Retrieval and Transfer System Description*

The functions of these systems are:

- Pre-selection and retrieval of the wastes
- Transfer the retrieved wastes to the conditioning building

During normal operation the equipment required for retrieval and transfer of the wastes are indicated in figure 3.2.1.

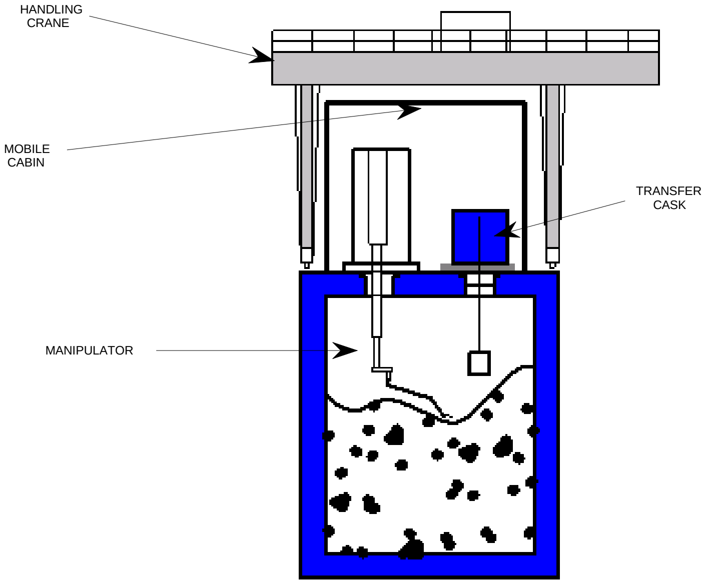


FIGURE 3.2.1 WASTES SELECTION AND RETRIEVAL

As a summary the equipment is:

- Waste retrieval manipulator
- Skip
- Transfer cask
- Mobile cabin
- Operculum sliding lids (gamma gates)
- Handling frame crane
- Other auxiliary equipment and structures to support the operation process.

The mobile cabin was designed as a steel structure to house, during the retrieval of the wastes, the manipulator and the transfer cask and provided to them protection against the action of the atmospheric elements.

The dimensions of the cabin allowed the embracing of two adjacent openings covered with sliding lids (gamma gates) to guarantee the silos confinement once its concrete plugs are removed.

The Manipulator and the transfer cask are positioned, through the blind type doors of the mobile cabin, over the corresponding operculum fitting them with its own gamma gates so that both act (open or close) simultaneously.

The handling of the equipment and elements was carried out by means of an overhead frame crane with a load capacity of 40 tons. This frame was the one used during the operation of the facility modified to update its machinery.

a) Main components of the retrieval manipulator (see figure 3.2.2):

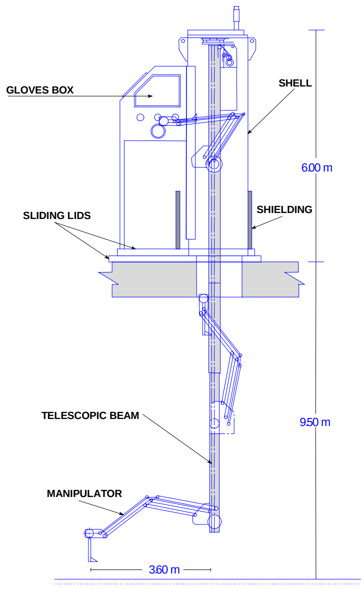


FIGURE 3.2.2 WASTE RETRIEVAL MANIPULATOR

Summarizing, the components of the retrieval manipulator are as follows:

- The telescopic beam, built with sliding elements, able to reach the bottom, including corners, of the silos.
- The manipulator arm endowed with rotating joints that allow a maximum reach, from the beam axe, of 3.6 m. A maximum loading capacity, under this condition, of 200 kg, including the weight of the tool accommodate at its end, was specified.

Two manipulators arms (adapted version of the Artisan manipulator from AEA Technology) were bought and the complete system was delivered in about 12 months).

- The tools to carry out the retrieval operations.

  Three types of tools were used:
  - Petal grab type tool able to grab and load bags, steel pieces and drums.
  - Spade type tool for pushing and loading
  - Rake type tool for rearranging, pulling and loading.
- The confinement shell provided the confinement of the silos during the retrieval operation and had to be shut, by means of its gamma gate during the transfer.
- A glove box, attached to the shell, provided the means for the replacement of the tools or even for the complete replacement of the manipulator arm.

  The manipulator was provided with lighting and video cameras (CCTV) to allow the remote control retrieval operation from an independent command cabin, located close to the mobile cabin.

b) Main components of the transfer cask (see figure 3.2.3):

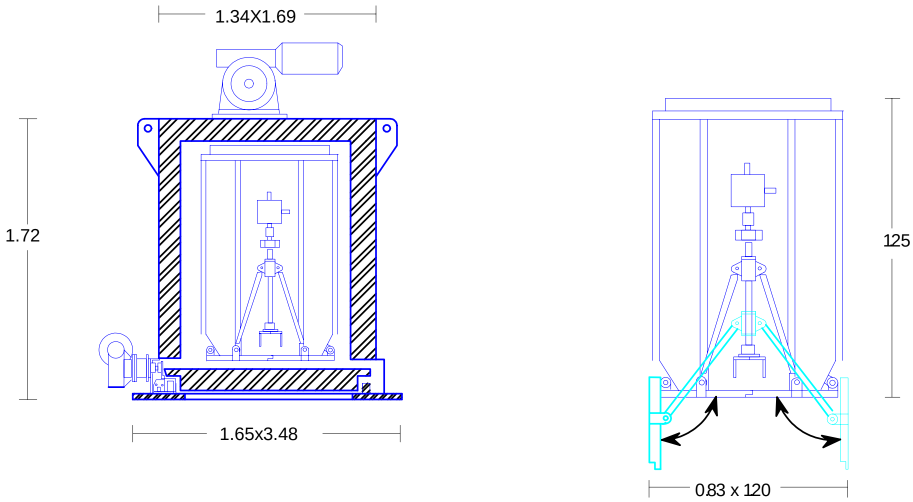


FIGURE 3.2.3 TRANSFER CASK AND SKIP

As a summary the components of the transfer cask are:

- The confinement shell with its gamma gate
- The skip
- The hoisting tackle for lowering or lifting the skip

Once the skip full, the opening and transfer cask lids had to be shut and the cask transferred from the silo operculum to the corresponding conditioning building transfer opening. As aforementioned the handling was carried out by means of the overhead frame crane.

Exceptionally, and due to the especial fullness conditions in silo 2, a specific initial extraction and transfer cask was designed to clear the space, under silo 2 openings, in order to achieve the require minimum room for normal retrieval operating conditions (about 1.8 m free under the roof slab).

The main components of the special cask are shown in figure 3.2.4 and they are:

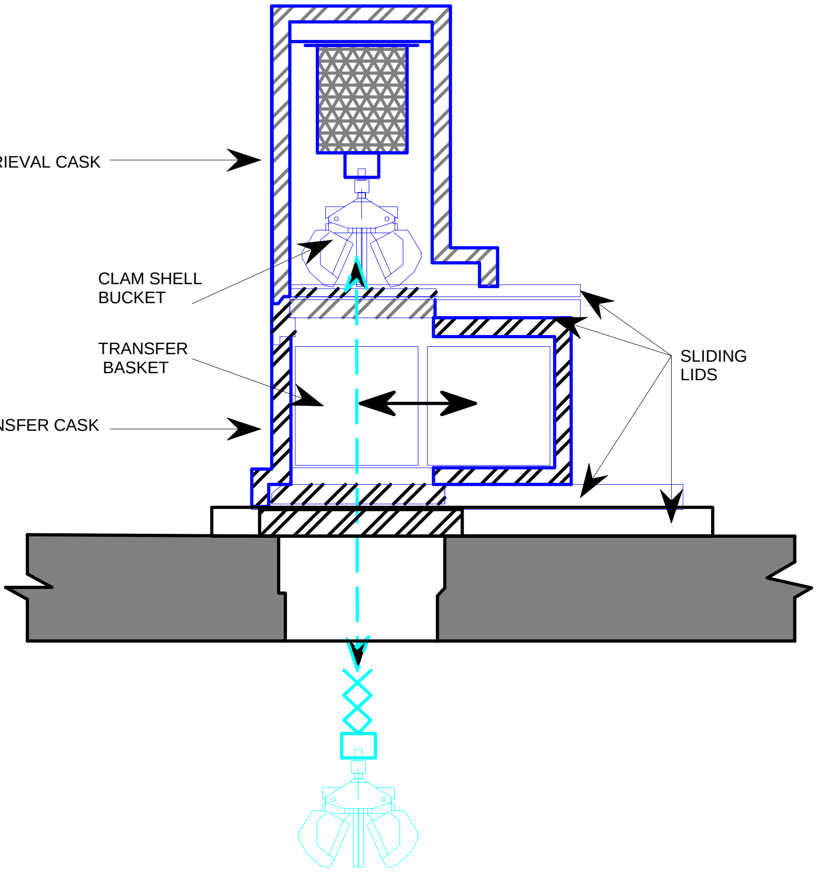


FIGURE 3.2.4 INITIAL EXTRACTION AND TRANFER CASKS FOR SILO 2

- Retrieval Cask
- Transfer Cask
- Clam Shell Bucket
- Sliding Lids (gamma gates)

Once the transfer cask be properly assembled to the operculum and the retrieval cask assembled to the transfer cask, the clam shell bucket proceed with the waste extraction and the loading of the transfer basket.

Once the basket full, the lids have to be shut and the cask disassembled so that the transfer cask is moved to the appropriate discharge opening, at the conditioning building, for wastes unloading.

As indicated in figure 3.1.1, and described in section 3.2, two different waste streams (diverse wastes and graphite/ absorbents wastes) had been foreseen and the corresponding transfer openings designed on the roof slab of the conditioning building.

3.3 Sorting and Conditioning Systems Description

The functions of these systems were:

- Sorting of the diverse wastes
- Processing of the graphite wastes and segregation of the support seat wires
- Packaging of the different stream of wastes
- Transfer of the waste packages outside of the conditioning building.

The process and transfer areas are housed inside a reinforced concrete building which main operating areas are indicated in figure 3.3.1.

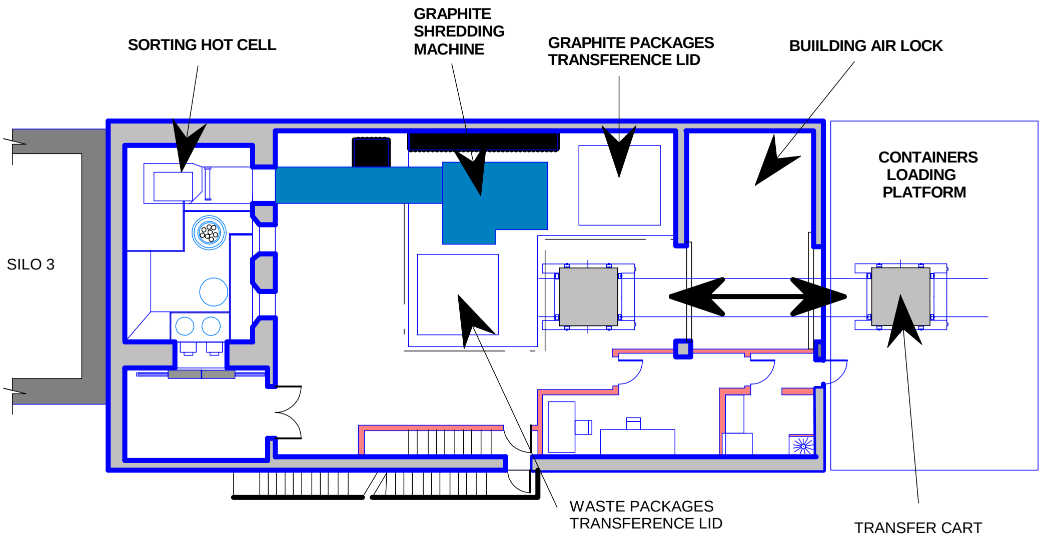


FIGURE 3.3.1 CONDITIONING BUILDING. OPERATING FLOOR

In the basement floor, below the operating floor, transfer corridors are arranged. The package transfer carts move the packages across transfer corridors, from the respective loading points to the corresponding transference area.

By means of an overhead crane, with a load capacity of 30 tons and an auxiliary hook of 2 tons, the package is lifted up to the operating floor and loaded on a transfer cart. The transfer cart moved the package outside of the building, passing through an air lock, where the package was to be taken by a fork lift truck and dispatched to the waste packages interim storage building.

The basement floor also lodges equipment belonging to auxiliary systems such as ventilation, fluids, compressed air and others.

Taking into account the type of wastes retrieved, the waste transfer cask are placed (see figure 3.3.1 and 3.3.2) over one of the two openings available in the sorting hot cell roof slab one for the graphite wastes stream an the other for diverse wastes stream.

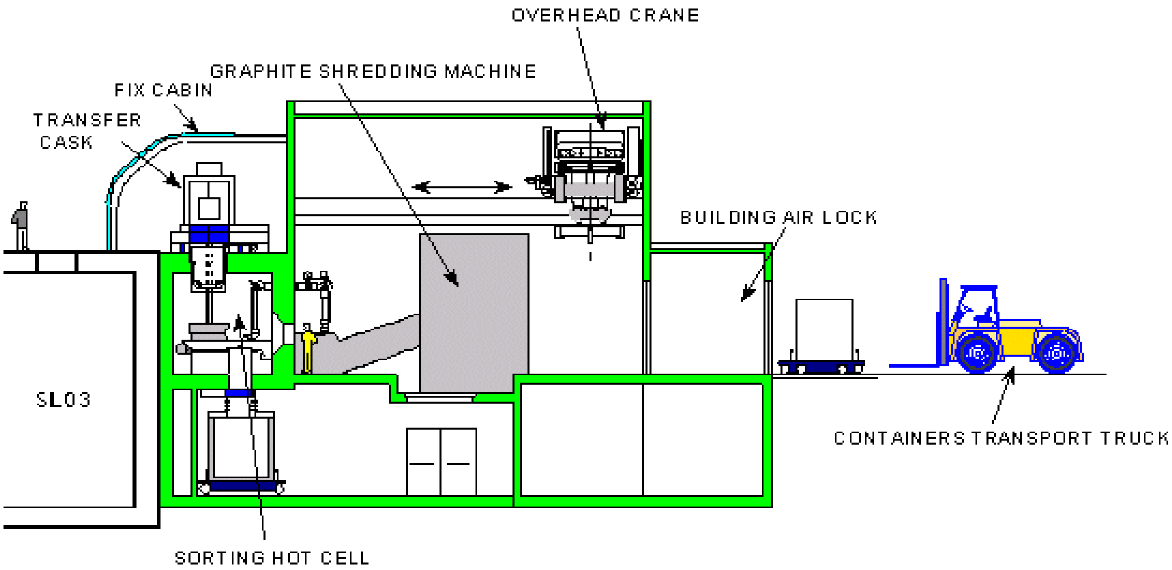


FIGURE 3.3.2 CONDITIONING BUILDING CROSS SECTION

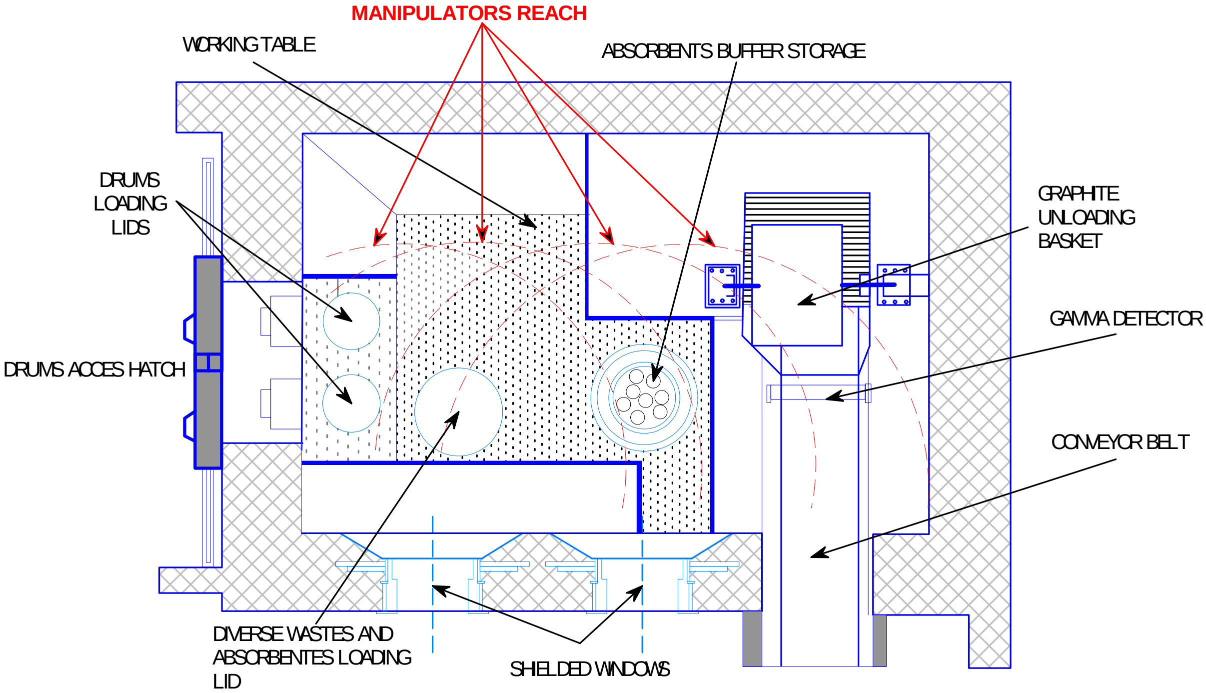


FIGURE 3.3.3 SORTING HOT CELL DETAILS

The hot cell main equipment and elements are (see figure 3.3.3):

- 2 shielded windows
- 4 wall manipulators
- 1 overhead lifting crane (500 kg)
- Working table below the diverse wastes unloading opening.
- Buffer storage for absorbents
- Basket elevator below the graphite wastes unloading opening.
- Conveyor belt to feed the graphite treatment line.
- Openings for the loading of waste packages:
    - 200 L drums for low level small size diverse wastes
    - CE-2 or CBE-1 (see figure 3.3.5) packages for other diverse wastes or absorbents.
- Lateral door for the access of the 200 L drums cart

The working process is as follows:

a. Diverse wastes

The skip from the transfer cask is lowered, through the roof slab opening, down to the working table level on which the wastes are unloaded.

By means of the manipulators, and eventually with the help of the lifting crane, the wastes are classified and loaded inside the corresponding waste packages (200 L drums or CE-2 package)

b. Graphite wastes

The skip from the transfer cask is unloaded in a basket elevator and, once lowered down to the graphite conveyor belt level, discharged the waste on it.

The conveyor belt feeds the graphite shredding machine (see figure 3.3.3) and is provided with a gamma detector that stop the belt in case of absorbent detection allowing the retrieval and buffer storage of the absorbents carried out by means of the wall manipulators.

Once full the absorbent buffer storage (9 absorbent units), they are loaded in a CBE-1 package through the same opening used for the CE-2 containers loading.

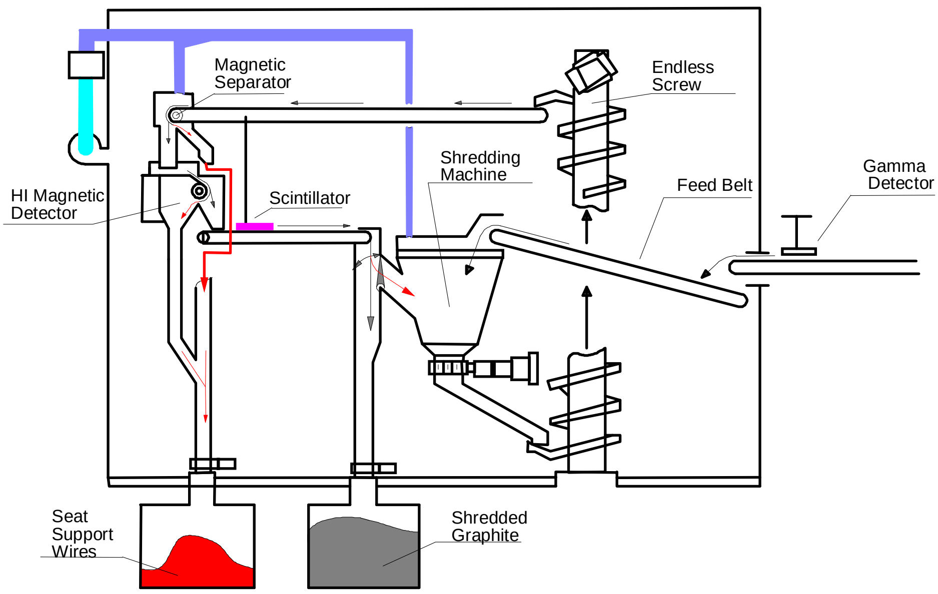


FIGURE 3.3.4 GRAPHITE CONDITIONING DIAGRAM

The graphite shredding machine main components are (see figure 3.3.4) as follows:

- Feeding Conveyor Belt
- Graphite Shredder
- Endless Screw
- Transfer conveyor belts
- Seat support wires detector and separator
- Bins and conduits to divert the steel support wires from the graphite and drive both of them to the corresponding containers (see figure 3.3.6):
  - CBE-1 for support wires
  - CME-1 for graphite
- Local ventilation system (Internal to the shredding machine)
- Machine envelope and gamma shield

The process flow diagram is shown in figure 3.3.4 and the arrangement and main dimensions of the components are shown in figure 3.3.5

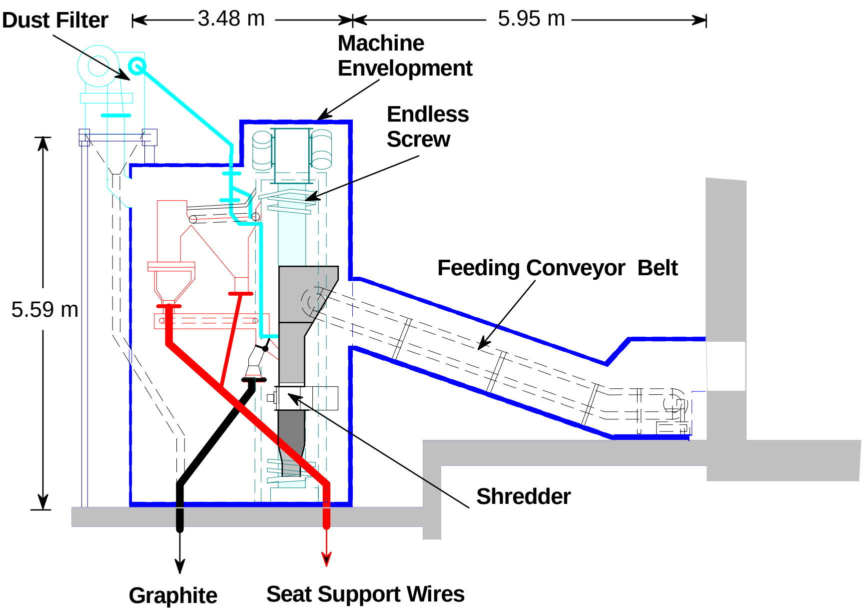


FIGURE 3.3.5 GRAPHITE SHREDDING MACHINE

For the machine dimensioning the following performance was considered:

- Machine graphite flow: 500 kg/h
    - Conveyor belt flow 350 kg/h
    - Recirculation flow 50 kg/h
    - Safety margin 100 kg/h

The estimated time for the treatment of the graphite contained in a skip (60 to 70 graphite sleeves units) is about 50 minutes.

In accordance with the results of the previous tests, the shredded graphite characteristics considered for the design were:

- Density: 1 gr/cm3
- Grain Size 1 to 3.5 cm

In addition to the 200 L drums used to package a certain type of diverse wastes, a summary of the waste packages, produced as result of the conditioning process, along with the type of waste packaged in them, is shown in figure 3.3.6.

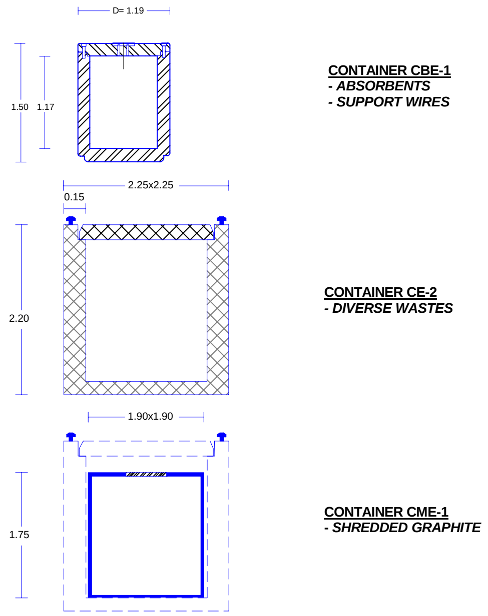


FIGURE 3.3.6 TYPE OF PACKAGES

3.4 Auxiliary Systems

3.4.1 General

The facility was equipped with the auxiliary system required to support the operation such as:

- Ventilation system
- Fire protection system
- Electrical system
- Fluid system
- Radiation Monitoring system
- Monitoring system
- Alarm system

When feasible, the systems were connected to the existing ones in the NPP.

Due to the characteristics of the wastes (mainly graphite), and for the purpose of the present report, only the ventilation system and the part of the fire protection system related with the fire extinction using Argon as extinguisher are outstanding.

3.4.2 Ventilation and fire protection systems

The system was designed to accomplish the following functions:

- Dynamic confinement of the radiological areas.
- Conditioning of the fresh air according with the requirements of each room.
- Air circulation from less to more contaminated areas.
- Air filtering before its release into the atmosphere.

The system was equipped with the appropriated measure, monitor, detection and control instrumentation required for system control and fire detection.

In case of fire in the silos or in the process areas of the conditioning building (Hot cell, graphite shredding machine and conveyor belt shell) the system was prepared to:

- Allow the injection of the extinguisher gas.
- Avoid an internal air pressure raise
- Maintain the confinement of the radiological areas.
- Cool the rooms ambient temperature
- Lower the temperature of the gases extracted.
- Restrict the effluents into the external environment.

To fulfill these functions the system was equipped as described below.

a) Silos ventilation and fire protection (see figure 3.4.2.1)

The silo fresh air is taken through and HEPA filter located at the manipulator shell.

The air is extracted through a two step filtration line, with filtering and extraction fan equipment, 100 % redundant.

The first step filtering box, pre-filter step, is equipped with a HE filter.

The second step box is equipped with two HEPA filters.

The ventilation equipment is lodged inside a mobile cabin that is handled with the overhead frame.

During retrieval operation the flow is equivalent to one renovation per hour of the silo air.

In case of fire, injection of Argon (gas) inside the silo is required. In order to fulfill the aforementioned functions the operating mode is as follows:

- Proceed with the Argon gas injection
- Maintain the air extraction
- Mix the extracted fumes with fresh air to lower the air temperature and protect the HEPA filters.

The optimum flow distribution (Argon, fresh air and extracted air) is achieved manually.

In case that the gas injection failed to extinguish the fire, it was foreseen a massive water injection into the silo.

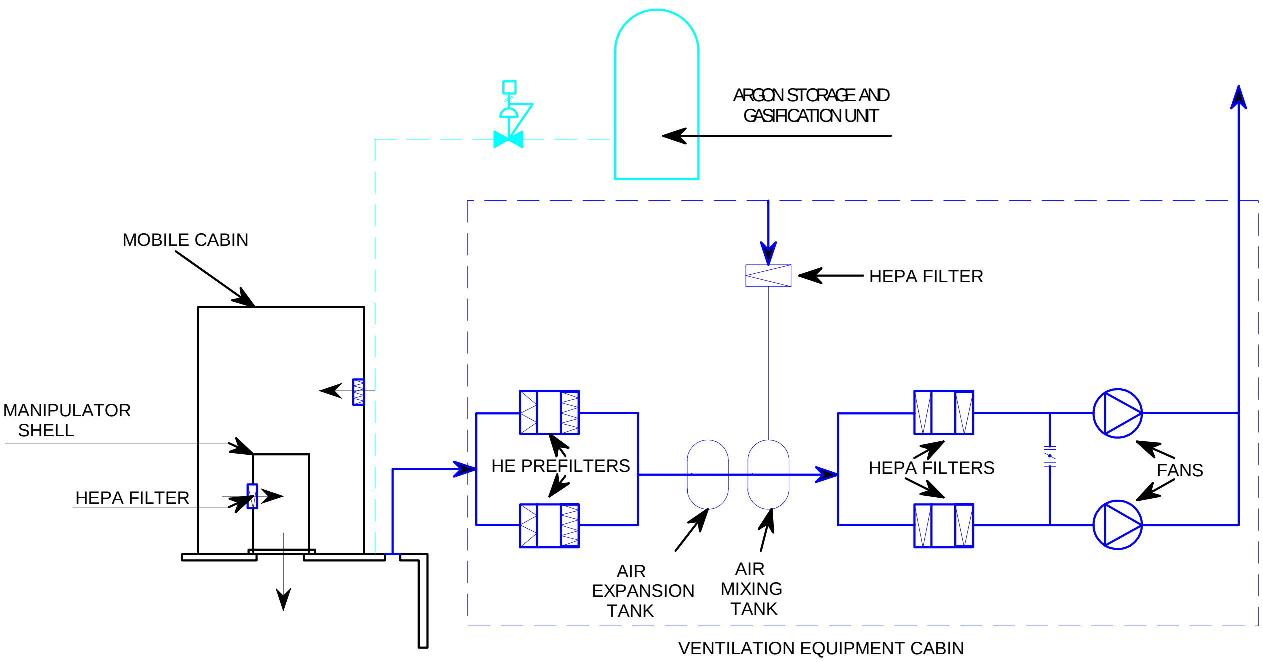


FIGURE 3.4.1.1 SILOS VENTILATION AND FIRE PROTECTION SYSTEMS

b) Conditioning building ventilation (see figure 3.4.2.2.)

The intake fresh air is conditioned and driven into the common areas of the building.

The air intake to the process areas are taken from the sorting hot cell lateral wall and from the containers loading areas and in both cases are equipped with HEPA filters.

The air is extracted through a two step filtration line, with filtering and extraction fan equipment 100 % redundant.

The first step is design with:

- HEPA filter for the air extracted from the common areas.
- Pre-filter and HEPA filter for the air extracted from the process areas.

The second step box is equipped with one HEPA filter.

The graphite machine was equipped with its own graphite dust removal HEPA filter operating in close loop.

During normal operation the ventilation flows considered were:

- 10 Volumes per hour in the hot cell.
- 10 Volumes per hour in the graphite machine (90 volumes per hour recirculation)
- 3 Volumes per hour in the building hall

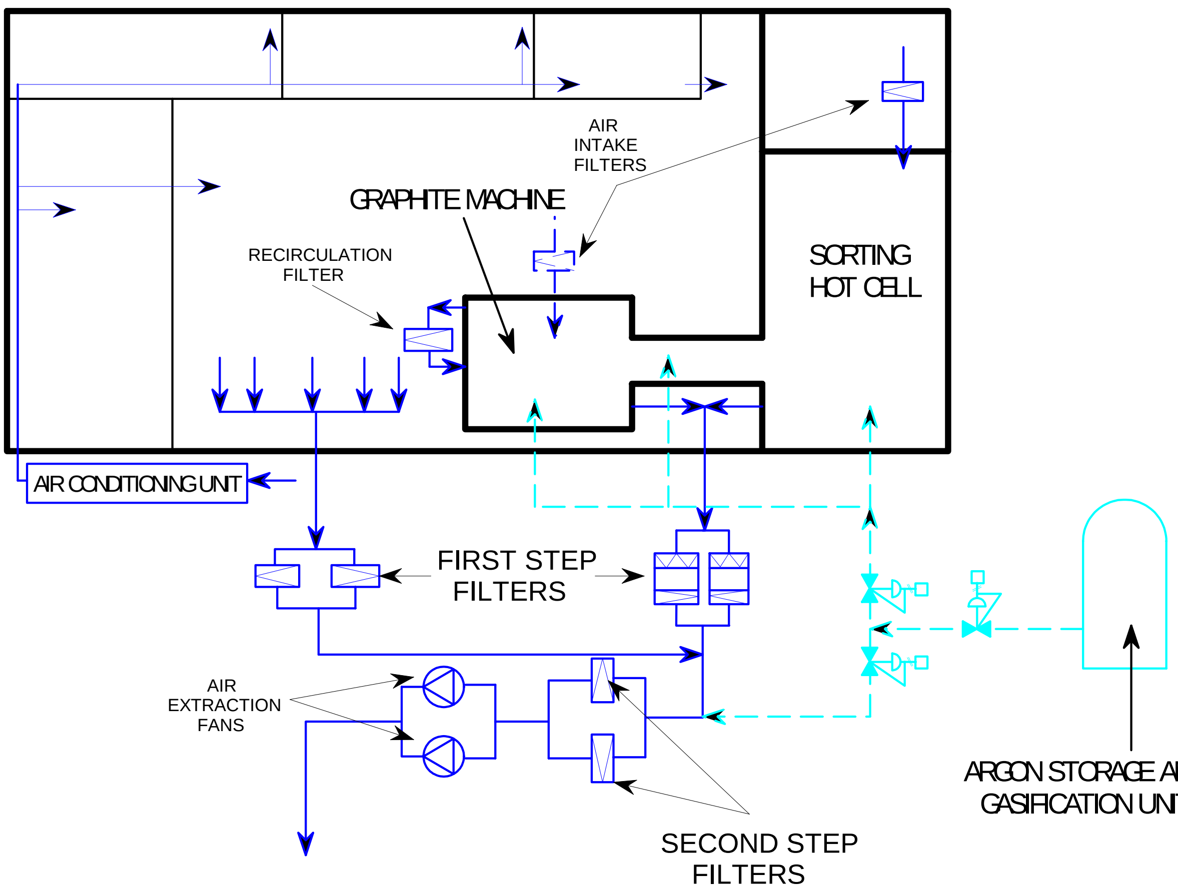


FIGURE 3.4.1.2 CONDITIONING BUILDING VENTILATION AND FIRE PROTECTION SYSTEMS

In case of fire inside the process areas (hot cell, graphite machine or conveyor belt rooms) the injection of Argon (gas) inside them is required. In order to fulfill the functions under this scenario the operating mode is as follows:

- The air intakes (hot cell and graphite machine) shut down.
- Stop of the process and the recirculation in the graphite shredding machine
- The non affected areas are isolated by means of the corresponding fire dampers
- Argon gas injection in the affected area.
- Maintain the area air extraction during a specified period of time. The flow is controlled to maintain the pressure conditions.

- Minimize the temperature of the extraction mixing the fumes with fresh air, coming from the hall, consequently protecting the HEPA filters.
- If the second step HEPA filter reaches a very high temperature, the area shall be isolated by means of the fire dampers shut down.

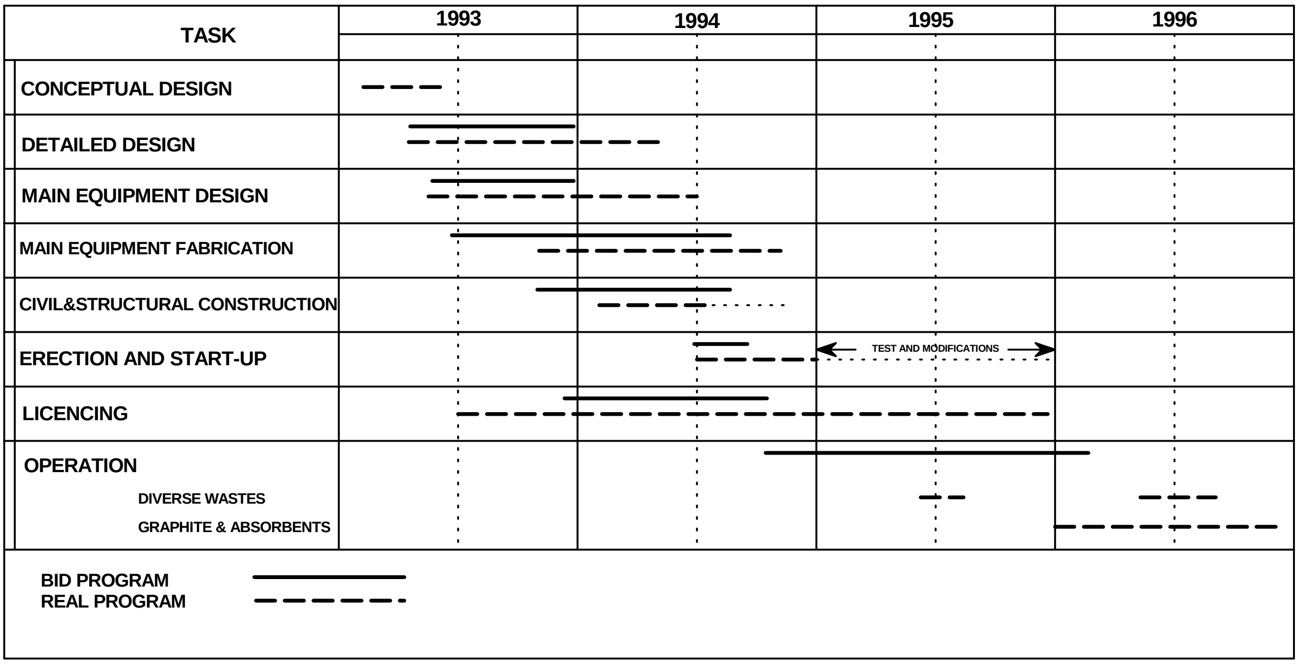
For dimensioning purposes the fire duration was calculated to last for about 15 minutes.

4. CONSTRUCTION PROGRAMME

The finalization of the activities related with the retrieval and conditioning of the radioactive wastes stored in the Vandellós I NPP Silos was the key condition for the transference of the facility ownership from HIFRENSA to ENRESA, this condition was present all along the project.

The Bid program foreseen the start of the project on September 1992 with a total project time of 19 months and 16 months of operating period. Due to issues arisen during the contract discussion and signature and the obtainment of the construction permits, the start of the project was delayed up to mars 1993.

It follows a general chronogram of the project showing the timing for the main task, as bided versus as carried out. For comparison purpose, the same start date, for the initiation of the detailed design, has been used for both chronograms.



5. OPERATING EXPERIENCES

Retrieval process started in Silo 3, that was filled only with graphite sleeves, and last approximately six months, with a stop of more than two months. Process continues in silo 2 for a 12 months period, and finished with retrieval of silo1 contents. Vacuum cleaning of inner walls and flours without any scabbing works, was performed after retrieval of wastes. Due to the penetration of contamination in concrete walls and floors (more than five cm depth in Cs-137 was reported), scabbing was required before demolition during NPP decommissioning works.

Total amount of graphite, steel wires and miscellaneous wastes was packaged in 232, CME type boxes (see figure 3.3.6) with 1054 t of graphite, 98 CBE type casks with 19,7 t of steel cylinders absorbents and wires, and 11 C2A type container with miscellaneous wastes, plus 875 drums with technological wastes generated during the retrieval process, that were sent to El Cabril LLW repository.

First CME was filled in December, 6, 1995 and last box was completed in January 22, 1998. During 1997, 114 boxes were generated and 116 in 1998, with an average rate of production of about 2 boxes a week.

Looking into daily production rates (see chart 1 and 2), two main standstills can be identified, one in April and May 1996, and other during June and July 1997.

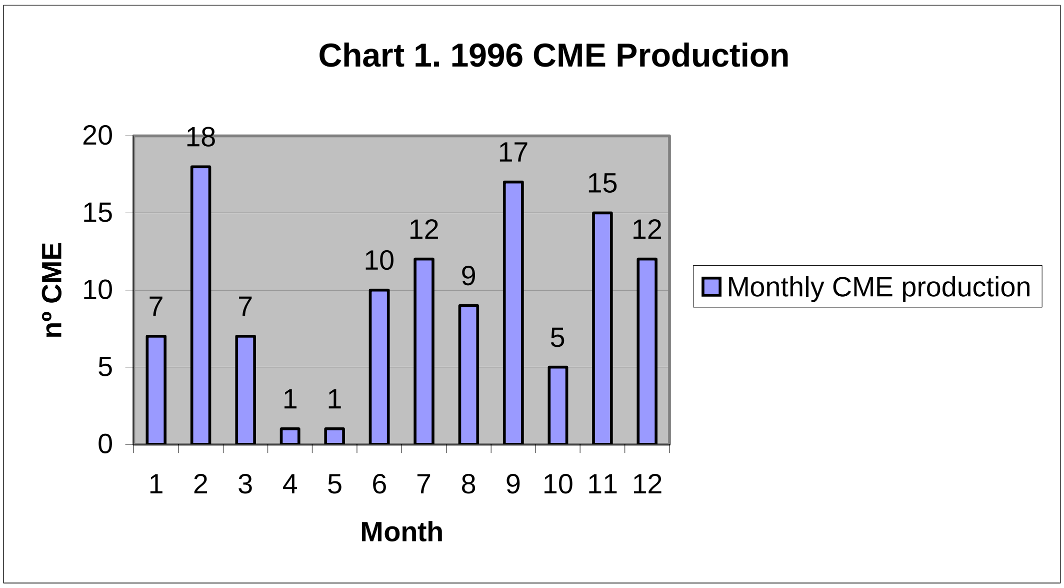


Figure 5.1. CHART 1. 1996 CME PRODUCTION

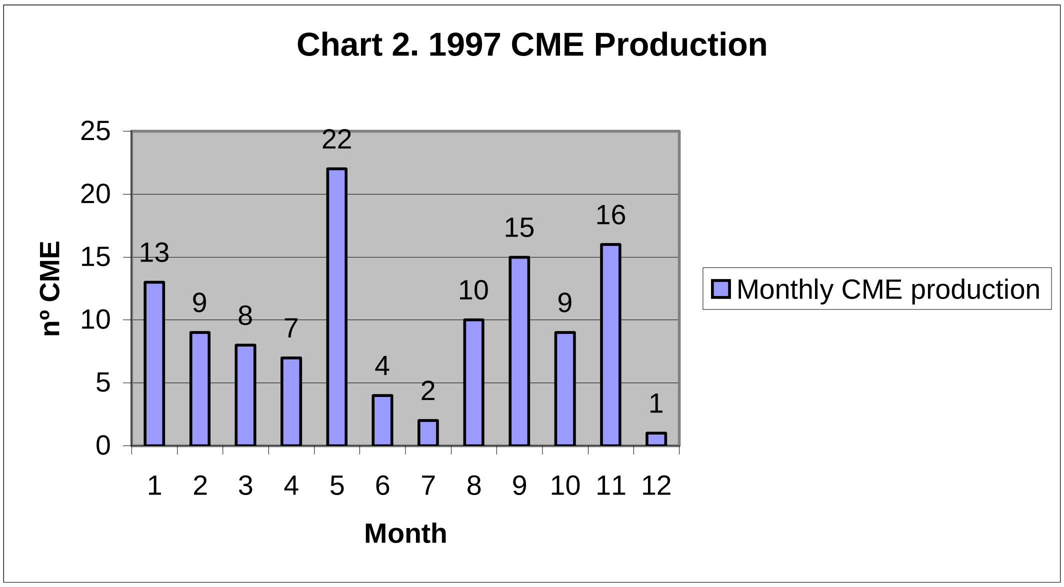


Figure 5.2. CHART 2. 1997 CME PRODUCTION

Crushing machine was reported broken down during May 1996, while in April some problems with spreaders of overhead crane were produced. To save time during this months, in April started retrieval activities of bags and miscellaneous wastes storage in Silo 1, with the aim of expedite future retrieval process of sleeves and absorbents in this vault.

In June 1996 a failure in the manipulator was reported during retrieval process in Silo 2. Due to the fact that this Vault was at this moment nearly filled, operations continued using petal grab with no significant impact in the Programme.

In June and July 1997, CME production falls down due to the change of all retrieval equipment from silo 2 to silo 1.

CBE casks were generated between December 26, 1995, and December 5, 1997 with an average rate of 1 cask a two weeks in 1996 and more that 1 cask a week in 1997.

According to daily production rates (Charts 3 and 4) low figures in 1996 are explained because of the absence of absorbents in Silo 3, therefore only steel wires with a very low volume were used to fill these casks.

To avoid large periods of maintenance in the identified weak points of equipment, a complete set of spare parts was available for the crushing machine and manipulator as a requirement of the client.

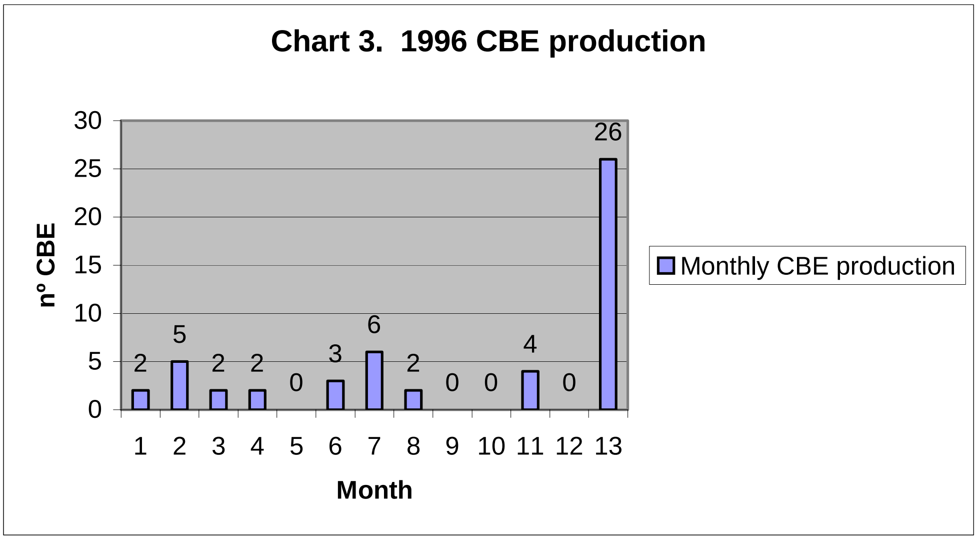


Figure 5.3. CHART 3. 1996 CBE PRODUCTION

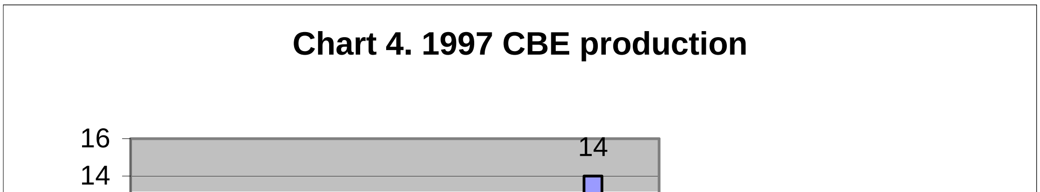


Figure 5.4. CHART 4. 1996 CBE PRODUCTION

In spite of different problems arise, mainly in the crushing cell, the impact of couple and uncoupled of all retrieval equipment from silo to silo seems to be more important due to all tests necessaries to compliance with isolation requirements established by the Nuclear Regulatory Authorities.

The average surface dose rate in CME boxes was: 3,66 mSv/h contact and 0,68 mSv/h at 1 meter. For CBE Casks, average surface close rate was 0,06 mSv/h, and 7,9 μSv/h at 1 meter.

In the case of the graphite CME boxes, the measured doses were higher than the designed doses (2 mSv/h contact dose and 0.1 mSv/h 1 meter dose). This was the only case in the project that, in relation to shielding calculations, had a discrepancy. The reason of such discrepancy can be found in the actually achieved rate of separation between graphite and steel support wires. This achieved rate caused an clearly higher than maximum designed amount of steel wires inside the graphite CME boxes (Designed maximum Graphite / Steel ratio in CME boxes: 80 / 20).

Due to high dose rate of CME boxes an overpack was built for transport from treatment building to short term interim storage building (see photographic report). This transport was made with a forklift truck having a lead glass for shielding purposes of drivers cab.

In May 1998 last CBE casks and C2A concrete containers were filled, with the final wastes resulting from operation.

APPENDIX 6 Retrieval and Segregation Characteristics and Properties

NOTE:

This area is under development in Work Package 1 is noted as reference 12. If appropriate an update will be included in the final WP2 report in 2011

H Eccles, NNL published this most recent version and it is recommended that NNL are consulted for the most up to date version of this document.

GRAPHITE CHARACTERISITCS / PROPERTIES PORTFOLIO - JUSTIFICATION

Introduction

At the WP 1 workshop (February 2009) participants considered the information being requested and its value to the CARBOWASTE project deliverables. It was appreciated that to obtain a comprehensive set of data would be both time consuming and arduous. To cover all reactors may also be impractical. It was agreed that a revised document would be distributed to WP1 participants and WP leaders. The revised document would require appropriate personnel to justify why the data/information were required, i.e. to justify why time should be expended obtaining it.

Information/data will only be retrieved for properties that are deemed of value.

Prof. Harry Eccles
March 2009

SPECIFIC

RETRIEVAL	
Reactor Vessel	**Justification**
Form/shape	
Material	
Dimensions (m)	
Reactor dimensions with shielding (m)	
Structural features in the reactor	
Access to containment structure	
Waste routes from reactor	
Core	
Shape and dimensions (m)	
Lattice	
Interaction with reactor structure	
Associated components and other materials	
Graphite block location	
Number of graphite blocks	
Clearance between blocks [pre and post operation] (mm)	
Other post operational clearances in reactor (mm)	
Post operation conditions	
Temperature (^0C)	
Humiidity (%RH)	
Graphite blocks	
Shape and dimensions (m)	
Weight [pre and post operation] (Kg)	
Post operation orientation	
Graphite properties [post operation]	
Tensile Strength	
Youngs modulus	
Poisson ratio	
Thermal conductivity	
Coefficient of thermal expansion	
Density	
Porosity	
Gas permeability	
Open pore volume	
Dynamic Young's Modulus	
Yield Stress	
Strain at Yielding	
Strain at Tensile Failure	
% Reduction in area at Failure	
% Elongation at Tensile Failure	
Shear stress	
Compressive stress	
Hardness	

SPECIFIC

TREATMENT	
Graphite properties [post retrieval]	**Justification**
Shape and dimensions (m or mm)	
Weight (Kg)	
Water content (%w/w)	
Oxidation rate in air/oxygen (ppm/^{0}C/hr)	
Electrical conductivity	
Thermal conductivity	
Density	
Porosity	
Gas permeability	
Open pore volume	
Shear stress	
Compressive stress	
Hardness	
Wetability	
Ash content	

SPECIFIC

RE-USE / RECYCLE	
Graphite properties	**Justification**
Shape and dimensions (m)	
Weight (Kg)	
Water content (%w/w)	
Oxidation rate in air/oxygen (ppm/^{0}C/hr)	
Electrical conductivity	
Thermal conductivity	
Density	
Porosity	
Radioactivity	

SPECIFIC

DISPOSAL	
Graphite properties	**Justification**
Shape and dimensions (m)	
Weight (Kg)	
Water content (%w/w)	
Oxidation rate in air/oxygen (ppm/^{0}C/hr)	
Density	
Porosity	
Gas permeability	
Open pore volume	
Shear stress	
Compressive stress	
Hardness	
Wettability	
Radioactivity	
Leachability of specific radionulcides	

APPENDIX 7 Groups and Factors influencing Retrieval and Segregation Criteria

Notes
This table is derived from a workshop open discussion ("Brain Storming) recorded on 11th November 2009 at NNL Risley. It has been rationalised in to subject areas and updated.

Political / Social / Economic
Corporate Policies (head office, site)
Cost and benefit (early or delay, investment priorities)
Disposal (routes, availability)
Dose impacts (early or delay)
Financial (planning, available funds)
Infrastructure (availability, modification)
International (practices, standards, public bench marks)
Legislation (current, changes, accelerate or delay)
Performance (throughput targets, failure rates, end points)
Political (national policy or law, interference, change of government)
Public (stakeholder definition, acceptability, local issues)
Regulatory (acceptability, change to standards)
Priority (hazard reduction, programme, window of opportunity)
Skill (competencies, availability, loss, qualification)
Technology (established, to be established, might improve, might be lost, transfer from other industries, manual versus remote)
Timing
Environment (changes, flooding, demography)
Finance (cost uncertainty, share costs with other projects, same of a kind sequence)
Knowledge (loss due to delay, data availability, resource availability)
Radiation Protection (dose impacts, impact of decay)
Public Acceptability (inconvenience, dose impacts, confidence)
Uncertainty (policy change, strategy change, resource lead times, resource availability, data availability, best available technology)
EHS&Q
Conventional Safety (Hazardous substances, heavy lifting, confined spaces, working at height, material and personnel movements, drowning, fire, asphyxiation)
Environment (external) (Best Practicable Means / Best Available Technology, emissions to water / land / air)
Injury (public, workforce)
Knowledge (operational experience, training, minimise complexity and training need, records past and future)
Process Engineering (in air, or underwater, shielding, reactor design, operational history, timing(decay), secondary waste, size reduction, in situ or not)

Radiation protection (public, workforce, radiation, contamination control, containment, transport, Wigner energy)
Regulatory (Safety case acceptability, safe systems of work, risk assessment)
Sustainability (minimise non-renewable resource usage)
Technology (established, available, maintainability, fault recovery, adaptability, robust in environment)
Waste (traceability, apply waste management hierarchy)
Waste Management
Minimisation (Secondary waste, primary waste packaging efficiency, recycle, reuse)
Packaging (Size, handling, classification for handling or transport)
Preliminary waste route (meets constraints, downstream conditions for acceptance including containers, stores, radiological, see also size, induced contamination, inherent contamination (solid, liquid, gaseous), secondary wastes, inherent)
Processes (Economic, mobile or fixed, primary waste, secondary waste, inherent waste, throughput, gross quantity, volume/density, working environment, flexible to improvements in technology/conditions for acceptance, simple, robust in environment, effective)
Segregation (compatibility with waste routes, re-categorise waste, meet conditions for acceptance, remove or mix gangue materials, recycle, secondary waste
Size Distribution (Graphite massive or particulate,
Technology (industrial applicability, can be modelled, predictable, suitable for environment, flexibility)
Reactor / Core design
Construction design (materials of core, pressure vessel, instrumentation, geometry of core, geometry of graphite blocks)
Graphite physical condition (solid or crumbling, weight, volume, weight loss, choice of tool, mechanical versus manual lifting (low activity situations), operational records)
Graphite radiochemical condition (irradiation history, impurities, activation products, circuit contamination, decommissioning contamination, operational records)
Infrastructure (cranes, access, ancillary operational areas,
Process (handle graphite, handle other materials, segregate graphite and other materials, mixed materials, efficiency of segregation, graphite in different roles and positions, graphite geometry
Size (throughput, reach all graphite, operational area footprints, channels, posting routes)
Space (access, headroom, craneage, depth and breadth of reactor,
Structural integrity (permissible loadings, structures, floors, cranes, all operational areas)

APPENDIX 8 - Comment on Anti Lifting Devices at Wylfa NPP.

Extract from email dated 13/07/09 from M Metcalfe to M Grave

In order to resist lifting forces on the bricks during fuel element extraction, the top 5 square bricks and the top 3 octagonal bricks are tied together in each fuel channel. The bricks are tied by magnox wire locked into a groove at the brick spigot.

A further refinement in the top layer is the presence of double length keys and keyways. The keys are trapped in the keyways to prevent ratcheting upwards during brick differential movements. The locking is achieved by 5/32 inch diameter steel locking wires above the keys between octagon bricks and blind keyways in the square bricks.

APPENDIX 9 – Hazard Identification and Operability Check List

Hazards and operability challenges may result from various operational requirements: -

General Graphite Retrieval and Segregation activity.

1. Visual inspection
2. Visual process viewing
3. Failure of equipment, e.g. electrical failure, spills of hydraulic oil
4. Break down, retrieval, maintenance of primary retrieval devices (access, handling and tools)
5. Bread down, retrieval, maintenance of secondary systems (viewing, ventilation, EC&I)
6. Tool change and replacement
7. Release of i-graphite blocks prior to handling
8. Attaching to the graphite
9. Varying strength of graphite (from powder to reasonable robust blocks)
10. Graphite in unexpected orientation or condition
11. Reach for tooling (commercial reactors are considerably larger than WAGR and GLEEP)
12. Blind spots (e.g. under corbels as experienced in WAGR)
13. Order of graphite removal
14. Positioning of main manipulating tool (e.g. RPV mounted machine or ROV operating on graphite stack or operational platform lowered over water (Bugey), reactor penetration as being considered for Ignalina
15. The effluent discharge/treatment associated plant for either in air or water or other environments (Note: specific HAZOP required for these plant but what are the needs with respect to the graphite e.g. dust management or water clarity)
16. Physical modelling (many conventional hazards in this list will be repeated during training of operators and trial on mock up rigs)
17. Buffer storage, how much and where, for what? (note: the whole retrieval process from handling through to dispatch requires to be scheduled to avoid critical path risks)
18. Dropped items on to the graphite and dropping of graphite
19. Layout of operational areas that support retrieval and segregation both in and out of containment
20. Classified areas and all other radiation and contamination risks
21. Avoidance of clashes between operators, tooling and waste
22. Maintaining containment
23. Damage caused to the reactor structure or surrounding buildings as a result of i-graphite removal operations (e.g. if high pressures or significant vibration result)
24. Radiation monitoring and sampling (e.g. i-graphite dust in instruments, cross-contamination of samples, risks to HP operators)
25. Sorting, size reduction, segregation of i-graphite and other materials including operation of waste packaging efficiency prior to dispatch

Unexpected issues: -

1. Graphite components in odd positions or stores e.g. mortuary tubes
2. Facilities located within the graphite core to facilitate non-reactor operations (note may be an issue with research reactors, for example GLEEP has a steel tunnel in the core used for irradiation experiments).
3. Understanding of graphite fire (see reference [10])
4. Wigner energy
5. Commercial constraints (consider gated process and risk management)
6. Contamination of working area and i-graphite by other materials not known to be present in the reactor containment or other associated working areas.

Omissions

The above lists are not intended to be complete; the HAZOP process should be commenced at the planning stage at upper level as discussed in Section 5.3.4 of the report and continued in more detail as the detailed process and design is generated for specific application. Other items may be identified on a case by case basis that are not in the above list. The above list is provided to seed the process and as a checklist. The use of HAZOP procedures that are recognised by the industry using trained personnel should be used.

APPENDIX 10 – Generic Processes for MCDA consideration

It is a presentation by David Ross at the Work Package 2 workshop and was issued as part of the workshop notes on SINTER.

National Nuclear Laboratory

Carbowaste WP2 Workshop

Gateshead, UK, July 14-16 2009

Presented by : David Ross

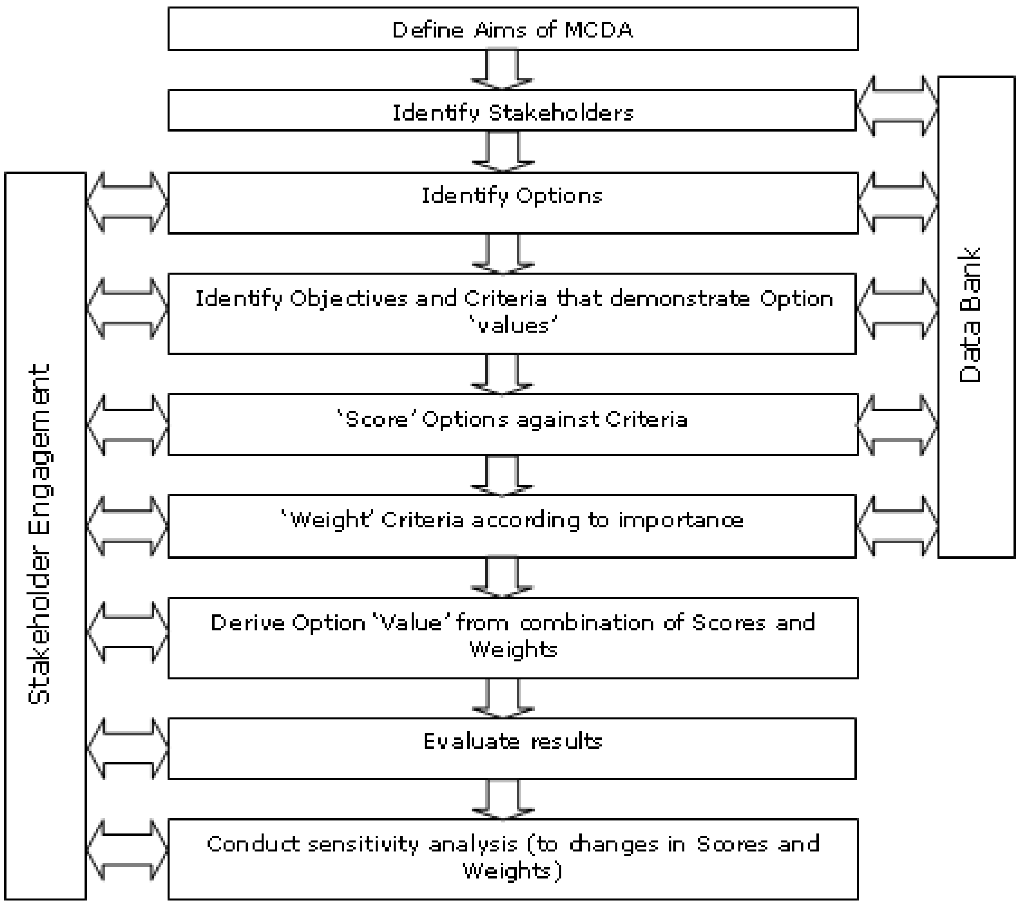
National Nuclear Laboratory

Date: 11/02/05

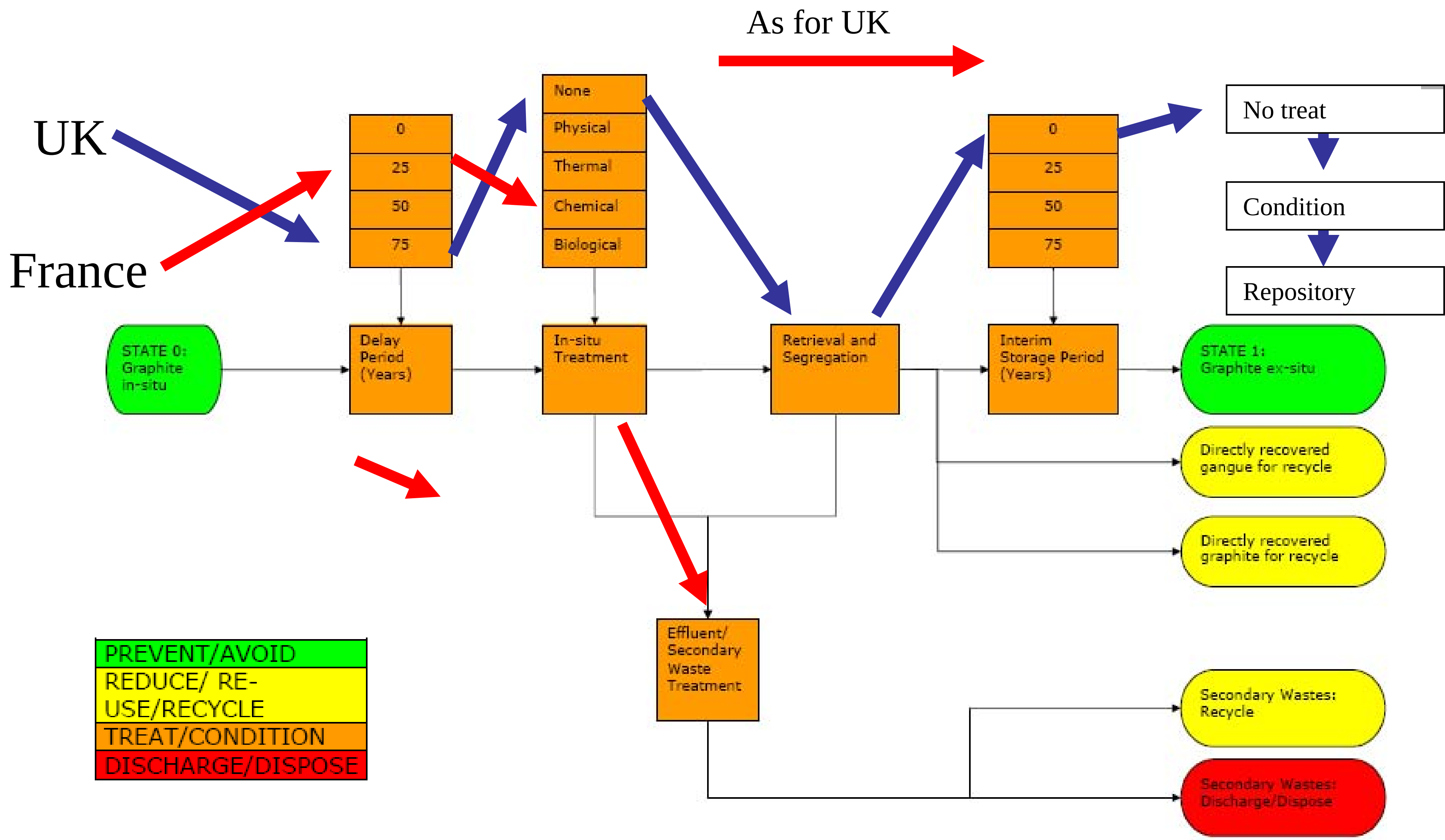
Workshop Rules

- Listen actively - respect others when they are talking.
- Only one discussion at a time.
- Do not be afraid to respectfully challenge one another by asking questions, but refrain from personal attacks.
- Participate to the fullest of your ability (but don't dominate the discussion) - share your own experience.
- Only ask questions during presentations if necessary to understand what is being said, leave all other questions to the end.
- Timely attendance and adherence to the Chairman/ Facilitator's direction in terms of timekeeping during sessions.

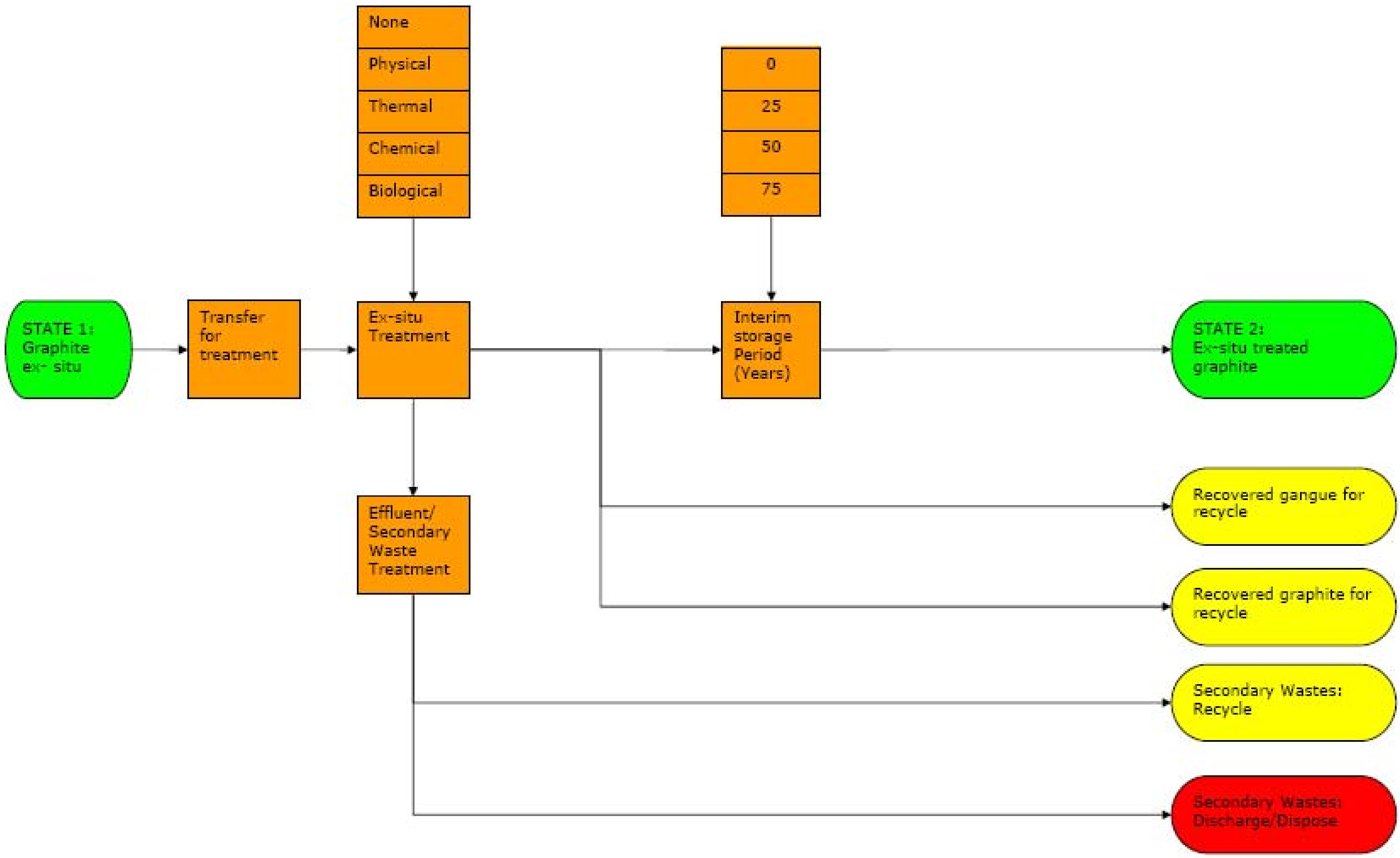
Generic MCDA Process



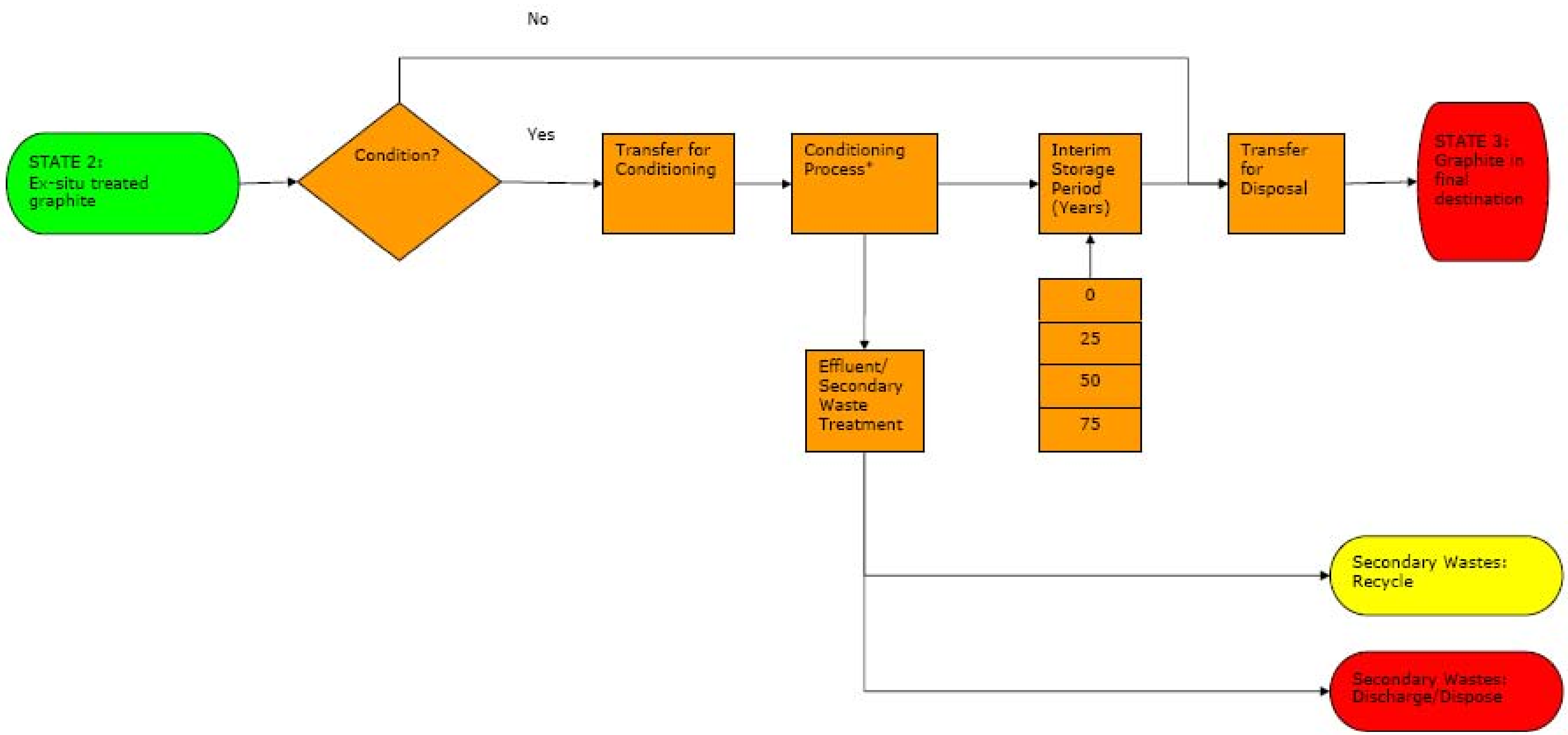
RETRIEVAL AND SEGREGATION



TREATMENT


None
Physical
Thermal
Chemical
Biological
0
25
50
75
STATE 1: Graphite ex- situ
Transfer for treatment
Ex-situ Treatment
Interim storage Period (Years)
STATE 2: Ex-situ treated graphite
Effluent/ Secondary Waste Treatment
Recovered gangue for recycle
Recovered graphite for recycle
Secondary Wastes: Recycle
Secondary Wastes: Discharge/Dispose

DISPOSAL



No
Yes
STATE 2:
Ex-situ treated graphite
Condition?
Transfer for Conditioning
Conditioning Process*
Interim Storage Period (Years)
0
25
50
75
Transfer for Disposal
STATE 3:
Graphite in final destination
Effluent/ Secondary Waste Treatment
Secondary Wastes: Recycle
Secondary Wastes: Discharge/Dispose

Generic Retrievals Process

PLAN

Initial core materials and reactor building characterisation, identify key retrievals criteria and develop plan for retrieval

RETRIEVE

Retrieve materials

POST-RETRIEVAL

Segregate; characterise wasteform and packaging; transport and interim storage and

Examples of Retrievals Strategies

- Intact under water
- Intact in air or inert atmosphere
- Fragmented in air or inert atmosphere
- Core routed to remove more active regions
- Combinations/variations on the above
- After 0, 25, 50, 75 years…

Examples of Segregation Strategies

- ‘Gangue’ segregated
- Graphite segregated at source into higher/lower activity (channel routing)
- Graphite segregated by reactor zone
- Graphite segregated based on activity measurement
- None
- ...

Examples of Treatment Strategies

- Chemical
- Thermal/Pyrolyse/Incinerate
- Mechanical
- Micro-biological
- Leaching
- None

Examples of Re-cycling and Re-use

- Re-cycle low activity graphite
- Re-use C14, H3

Disposal Strategies

- Surface
- Shallow
- Deep

Disposal may be to a dedicated facility for reactor graphite, or could involve co-disposal with other radioactive materials i.e. LLW, ILW or HLW.