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Treatment and Disposal of Irradiated Graphite and Other Carbonaceous Waste

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WP 1 Review Report - Introduction

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Integrated Waste Management Approach WP 1 Review Report - Introduction

Executive summary

This document is the introduction to the Carbowaste Work Package 1 Review Report. The summary report consists of a series of chapters that address the i-graphite position in various Member State and other countries that have constructed/operated graphite moderated reactors. In addition a specific chapter addresses the current disposal of i-graphite and factors that need consideration for final disposal of this waste. As treatment technology to remove C-14 could give rise to the liberation of carbon dioxide, capture and sequestration (carbon capture) are also considered.

This document will evolve as the CARBOWASTE project progresses and will be reissued at key points of the programme.

Revisions

Rev.	Date	Short description	Author	Internal Review	Task Leader	WP Leader
01	23/12/08	1	AW Banford (UK NNL)	H Eccles (UK NNL)	H Eccles (UK NNL)	H Eccles (UK NNL)
02	04/03/09	Consistency of format checked	AW Banford (UK NNL)	H Eccles (UK NNL)	H Eccles (UK NNL)	H Eccles (UK NNL)

Contents

1 Introduction

This report has been prepared as part of a European collaborative project "Carbowaste Treatment and Disposal of Irradiated Graphite and Other Carbonaceous Waste". Specifically, it is the Summary Report of Work Package1, Task 1.1, and includes individual chapters prepared by the member organisations involved in this task.

The aim of the report is to give an overview of the current situation regarding irradiated graphite internationally. This is a "living document" that will evolve as the Carbowaste project progresses and will be reissued at key points in the programme.

1.1 The Carbowaste Consortium and Project

A four year collaborative European Project 'Treatment and Disposal of Irradiated Graphite and other Carbonaceous Waste (Carbowaste)' was launched in May 2008 under the 7th EURATOM Framework Programme. The aim of the project is to develop best practices in the retrieval, treatment and disposal of irradiated graphite, addressing both existing legacy waste as well as waste from graphite-based nuclear fuel resulting from a new generation of nuclear reactors (e.g. V/HTR). The consortium is led by Forschungszentrum, Juelich (FZJ) in Germany and involves 28 partners from Belgium, France, Germany, Italy, Lithuania, Netherlands, Romania, South Africa, Spain, Sweden and the United Kingdom.

1.2 The European graphite legacy

Across the world, graphite has been used extensively as a reflector and moderator material in over a hundred nuclear power plants, research reactors and plutonium production reactors. Indeed, according to the IAEA[1], there are over 230 000 tonnes of irradiated graphite which will ultimately need to be managed, potentially as radioactive waste. A significant portion of this irradiated graphite is located in Europe.

In the UK, for example, there is a large legacy of irradiated graphite as a result of its use in graphite-moderated gas-cooled reactors, the Magnox reactors and the Advanced Gas-cooled Reactors (AGR). Various European countries and others around the world also have legacy

irradiated graphite. The reactors vary in size and type across Europe and include research/test reactors as well as larger base-load reactors; such as the RBMK at Ignalina in Lithuania, and the French gas-cooled reactors. Whilst some graphite has already been removed and packaged, as in the case of the Graphite Low Energy Experimental Pile (GLEEP) and Windscale AGR (WAGR) in the UK, on other sites the graphite remains in the reactor building.

1.3 The Carbowaste context of this review document

This review document should be viewed in the context of the overall work of Work Package 1. This document presents an overview of the current situation with regard to the status of irradiated graphite across Europe. Other Work Package 1 activities are progressing in parallel, notably the development of a route map to identify the key issues associated with the management of irradiated graphite, and the development of graphite data sheets for each country.

In compiling the information and data, the authors of each chapter have endeavoured to collate and capture all key references. Many other nuclear graphite references are available but have not been included as they focus on other aspects of graphite behaviour, but are less relevant in the context of irradiated graphite waste management and/or retrieval of i-graphite.
This review document, the route map and graphite data will be reviewed subsequently to identify the information required to inform waste management strategy and identify gaps. The output of this exercise will inform the work of the Carbowaste consortium in future years.
Although the retrieval and possible treatment of irradiated graphite is a key component of the decommissioning of graphite moderated reactors, the decommissioning considerations have been limited to those that will directly affect the waste management of i-graphite.

1.4 Structure of the Review Document

The review document comprises a range of chapters prepared separately by the work package participants as detailed in Table 1. The authors were selected either for:

1 knowledge of their own reactor situation and that of a neighbouring Member State;
2 knowledge of their own reactor position and that in other countries;
3 work commitment to Work Package 1 and this specific task.

Where ever possible, the information in these chapters has been reviewed by an independent reviewer.

Table 1: Work Package 1 Review Document

Chapter	**Prepared by**	**Chapter**	**Prepared by**
Introduction	NNL		
Overview of Nuclear Graphite	NNL		
European Chapters		*Non – European Chapters*	
France	EDF	Asia	NNL
Germany	FZJ	Japan	NNL
Italy and Spain	ENRESA	South Africa	PBMR
Lithuanian	FI	US	NNL
Russia	LEI		
UK	NNL		
Other Member States	NNL & FI	*Cross Cutting Chapters*	
		Disposal	ANDRA
		CO_2 sequestration	Doosan Babcock

The structure of the report allows the individual country specific chapters to be read 'stand alone'. However, it is recommended that the reader first considers this 'Introduction' chapter and the 'Overview of Nuclear Graphite' chapter.

2 The Irradiated Graphite Challenge

The challenge associated with the integrated waste management of irradiated graphite is summarised in Figure 1. Nuclear graphite is first manufactured and formed into components for nuclear applications, such as graphite bricks, fuel sleeves and fuel compacts. The graphite component formed does however contain impurities. Once in a reactor, the graphite components are exposed to radiation. As a result of reactor operation, these components will potentially have associated activation products from manufacturing impurities and through contamination by transported material in the circuits. The presence of these radioactive materials then complicates any future operations, be it retrieval, treatment, storage, disposal or recycle/reuse.

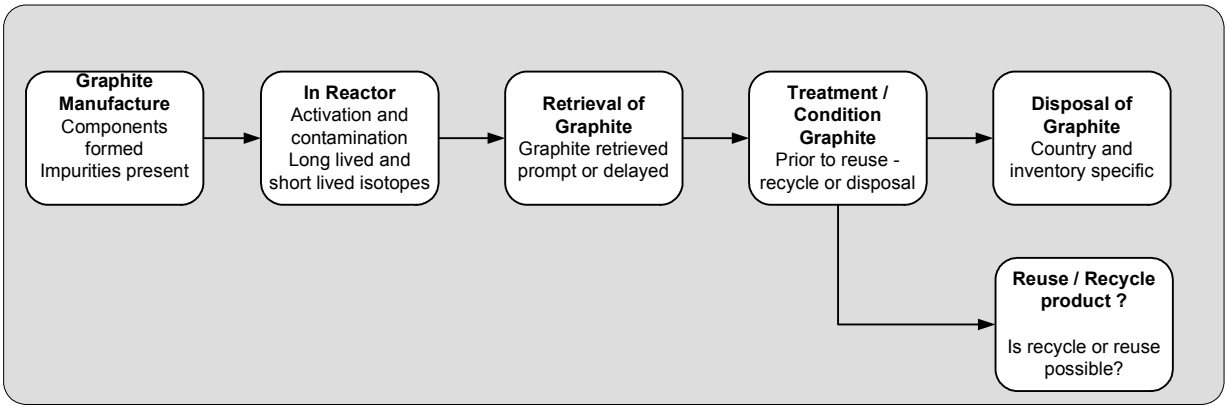

Figure 1: Irradiated graphite management

2.1 Graphite Inventory

As stated above, the IAEA estimated the total quantity of irradiated graphite worldwide at over 230, 000tonnes. This graphite has been used in range of nuclear reactors from the large power-producing reactors such as the gas-cooled reactors in France, the UK's Magnox and AGR stations, and the Russian-built RBMKs. Furthermore, graphite has been used extensively in a number of plutonium production reactors, research reactors, Material Testing Reactors (MTRs) and prototype/experimental reactors (for example, High Temperature Reactors (HTRs)). The graphite is in range of forms and geometries, based on the reactor and fuel design. Within Europe, the largest quantities of graphite are associated with UK, French and Russian design reactors.

2.2 Retrieval of Graphite

Graphite has been retrieved from a number of reactors world-wide. The most significant experience to date has been gained decommissioning Fort St Vrain in the US, GLEEP at Harwell in the UK and WAGR at Windscale also in the UK. The reactor at Fort St Vrain was a HGTR design and the graphite was removed under water, to provide shielding for the dismantling operations. An alternative approach was used in the UK with retrieval of graphite from WAGR being carried out dry but remotely. The GLEEP reactor which was of a smaller size and lower radiation dose was also dismantled in air and conditions allowed a more manual approach to be adopted.

The current position regarding dismantling of reactors varies both from country to country and from reactor to reactor. The current contrasting baselines in the UK and France illustrate the point. The current UK strategy delays reactor decommissioning to take advantage of the decay of short lived isotopes, reducing the complexity of remote operations whereas the strategy in France is for prompt (relative to the UK strategy) decommissioning.
The retrieval and segregation of graphite including HTR fuel coatings is considered in detail in Carbowaste Work Package 2

2.3 Treatment of Graphite

The treatment of irradiated graphite may be desirable for many reasons;

- to reduce dose to workers,
- to reduce volumes of waste for disposal,
- to achieve the radiological requirements for disposal,
- to facilitate recycle or reuse of isotopes or graphite.

A wide range of treatments has historically been considered[1]. These range from immobilisation to incineration.

Immobilisation using a range of encapsulants, including cement, polymer, resin, sand, bitumen, glass, metals and ceramics has been considered and evaluated for the disposal of graphite. In the UK, the WAGR graphite has been immobilised in cement and is now stored awaiting disposal. A similar immobilisation approach is planned for graphite when retrieved from Bugey (France).

Thermal processes have also been widely investigated for the treatment of irradiated graphite, ranging from furnace incineration to steam pyrolysis. The goal of such an approach would be a dramatic reduction in waste volume. However, the associated challenges of separation of radioactive gases, disposal of ash and the difficulty of burning graphite[1] must also be noted.

The Carbowaste project is considering the treatment of graphite in Work Package 4; specifically consideration will be given to physical, chemical and microbiological processes.

2.4 Recycle – Reuse

Recycling and reuse of graphite is addressed specifically in Work Package 5. A review of existing experience relevant to recycling will be included in this work package.

It is however worth noting that recycling of material (either graphite or separated isotopes) will depend heavily on demand and cost. Notwithstanding this, material has been recycled from the nuclear industry into mainstream use, one example being the reuse of aluminium recovered from the redundant Capenhurst gaseous diffusion plant.

2.5 Disposal

Options for the disposal of irradiated graphite are influenced by many factors including radioactive inventory, volume, waste form, timing and not least by the availability of a suitable storage facility with its own associated waste acceptance criteria. The aims of any pre-treatment and conditioning of waste must be consistent with the disposal requirements. This is an area where local infrastructure, policy and regulation vary from country to country. The disposal chapter prepared by ANDRA considers disposal internationally as a single theme. Work Package 6 within the Carbowaste project will subsequently address issues associated with disposal.

3 Summary

This introductory chapter has given a high level overview of some of the issues associated with the management of irradiated graphite waste in the context of the Carbowaste project. The subsequent chapters give a more detailed view of nuclear graphite and its management from both generic and country-specific perspectives.

4 References

1. [1] IAEA , “Characterization, Treatment and Conditioning of Radioactive Graphite from Decommissioning of Nuclear Reactors, IAEA-TECDOC-1521, IAEA Austria, (Sept 2006)