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Integrated Waste Management Approach WP 1 Review Report - USA Chapter

Executive summary

This chapter addresses the position with respect to US legacy waste irradiated graphite (i-graphite). There are 34 nuclear reactors listed in the Department of Energy (DOE) inventory as having graphite incorporated into the design. However, only twelve of these reactors contain (or contained) significant volumes of graphite: the Hanford production reactors, the Brookhaven Graphite Research Reactor, the Peach Bottom experimental high temperature gas reactor and the decommissioned Fort St Vrain high temperature gas cooled reactor. The amount of i-graphite in the US is estimated to be very approximately 15,000 tonnes of graphite based upon volumes of material from these sites.

There are two approaches to decommissioning in the US, depending upon whether the nuclear plant are within the commercial sector licensed by the Nuclear Regulatory Commission (NRC) or are managed by the US DOE. In the case of plant under the jurisdiction of the NRC, three methods for decommissioning are available : DECON (immediate dismantlement), SAFSTOR (facility maintained to allow decay of radioactivity, later followed by dismantlement) and ENTOMB (entombment and monitoring). The DOE approach covers transition, deactivation, surveillance and maintenance and decommissioning.

Decommissioning of US nuclear reactors containing graphite includes the dismantlement of the Fort St Vrain plant under DECON, the dismantlement of the SR 305-M Test Pile at Hanford (DOE jurisdiction) and the decommissioning and burial on site of the CP-2 (formerly Chicago Pile 1 and under DOE jurisdiction). The graphite from the fully decommissioned Fort St Vrain reactor is now stored on a DOE site. Decommissioning of the Brookhaven Graphite Research Reactor is currently in progress. At present, the US has no plans for the treatment of irradiated graphite with disposal as Low Level Waste on DOE sites being the favoured approach.

Revisions

Rev.	Date	Short description	Author	Internal Review	Task Leader	WP Leader
01	23/12/08	Issue for WP review	MP Metcalfe (UK NNL)		H Eccles (UK NNL)	H Eccles (UK NNL)

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Chapter contents

1 Introduction

This chapter addresses the position with respect to US legacy waste irradiated graphite (i-graphite). This is estimated to be very approximately 16,000 tonnes of graphite based upon volumes of material in the Hanford reactors and at the Brookhaven, Peach Bottom and Fort St Vrain sites.

US nuclear reactors as described by the US Department of Energy (DOE)[1] fall into two categories : civilian (commercial) reactors and production reactors. Civilian reactors containing i-graphite fall into a number of sub-groups : shutdown power reactors (two), shutdown experimental power reactors (two), operable university reactors (one), shutdown university reactors (five), operable research and test reactors (one), shutdown research and test reactors (eleven). Production reactors containing i-graphite fall into two sub-groups : shutdown materials production reactors (nine), shutdown process development reactors (three). Operable production reactors are excluded from the DOE review as this information is classified. A more detailed breakdown of the type of nuclear reactor containing i-graphite and its status will be provided below. Any reactors containing graphite commissioned post-2003 are excluded from the US legacy waste inventory. For the purposes of this review, all graphite identified in the 2003 DOE list has been classified as i-graphite. However, it should be noted that some of these reactors have already been decommissioned.

2 The Legislative and Regulatory Framework

The US ratified the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management (hereafter referred to as the Joint Convention) in April 2003. The Joint Convention establishes an international peer review process among contracting parties and provides incentives for nations to take appropriate steps to bring their nuclear activities into compliance with general safety standards and practices.

A second US national report[2] for the Joint Convention documents spent fuel and radioactive waste management safety. This report, which is currently being revised and updated as the third national report, describes radioactive waste management in the US in both commercial and government sectors, providing annexes with information on spent fuel and waste management facilities, inventories and ongoing decommissioning projects. This second national report is the primary source for the policies and practices summarised below.

2.1 US National Policy on Nuclear Activities

The US Nuclear Regulatory Commission (NRC) began regulatory operations in 1975 and performs its mission by issuing regulations, licensing commercial nuclear reactor construction and operation, licensing the possession of and use of nuclear materials and wastes, safegauarding nuclear materials and facilities from theft and radiological sabotage, inspecting nuclear facilities and enforcing regulations. NRC regulates commercial nuclear fuel cycle materials and facilities. NRC is also responsible for licensing commercial nuclear waste management facilities, independent spent fuel management facilities and the planned Yucca Mountain repository for disposal of high-level waste (HLW) and spent fuel.

The DOE was made responsible for nuclear energy technology and nuclear weapon programmes. The DOE has added new nuclear-related activities for environmental clean-up of contaminated sites and surplus facilities. The DOE is responsible for developing the planned Yucca Mountain site as a repository.

The US Environmental Protection Agency (EPA) was created in 1970 to address a growing public demand in the US for cleaner water, air and land. The EPA was assigned the task of repairing damage already done to the environment. The EPA established generally applicable environmental standards for the protection of the general environment from radioactive material. Among its many responsibilities, the EPA also establishes standards for management and disposal of spent fuel, HLW and transuranic (TRU) waste.

2.2 The Government Sector

DOE is responsible for and performs most of the spent fuel and radioactive waste management activities for government-owned and generated waste and materials located, for the most part, on government-owned sites. These activities include management of spent fuel remaining from defense reactor operations, primarily at the Hanford Site, Washington and Savannah River Site, South Carolina. These operations ceased in the early 1990s. DOE also provides safe storage for the core of the decommissioned (civilian) Fort St Vrain high temperature gas reactor (HTGR).

DOE has a complete waste management system for government spent fuel and waste. This includes numerous storage facilties and processing facilities (treatment and conditioning).

2.3 The Commercial Sector

Owners and operators of nuclear power plants and other types of facilities generating radioactive waste manage the spent fuel and radioactive waste generated by the facilities prior to disposal. Waste disposal sites will ultimately be administered by the US Federal or state governments. Government custody may occur at different stages of the waste management scheme depending upon the type of radioactive waste and activity.

2.4 Classification of Radioactive Waste

The US radioactive waste classification system has two separate subsystems, one applying to commercial and the other to DOE waste.

Radioactive waste from DOE nuclear operations is classified as HLW, TRU waste, LLW or mill tailings. Waste may also contain hazardous waste constituents. Waste with both radioactive and hazardous constituents in the US is called ‘mixed’ waste.

LLW is classified in the commercial sector as Class A, Class B, Class C and Greater-than-Class C (GTCC) LLW. These classes are defined in NRC regulations with concentrations of radioactive material increasing from Class A through to GTCC.

DOE manages waste from its operations using procedures and requirements comparaable to those used by NRC for commercial waste. Both NRC and DOE approaches apply similar performance objectives.

Cross-reference to the IAEA waste classification scheme is approximate based on available waste management data and the US has provided information defining its classification system and comparing it with proposed IAEA waste classes to the Net-Enabled Waste Management Data Base programme at the IAEA in 2004.

2.5 Radioactive Waste Management Practices and Decommissioning

Radioactivity in LLW can cover a broad range, from just above background (such as some protective clothing) to very high levels (such as parts from inside the reactor vessel in a nuclear power plant). The US has a comprehensive system for most LLW, with commercial and government facilities existing for LLW processing, including treatment, conditioning and disposal. Generators prepare LLW for shipment to licensed disposal. Large volumes of LLW have been generated in recent years from facility decommissioning and site remediation. LLW (Class A, B and C) is currently disposed in near surface facilities.

Classification as 'TRU waste' exists only within DOE government (non-commercial) sector. The principal repository for TRU waste is the Waste Isolation Pilot Plant in New Mexico. The planned Yucca Mountain repository, if licensed, will be used for the disposal of HLW.

Decommissioning is an activity generally taking place at the end of operation of commercial and government nuclear facilities. Waste from decommissioning is managed within the waste classes briefly described above.

Radioactive wastes are treated primarily to produce a structurally stable final waste form and minimise the release of radioactive and hazardous components. The US does not commonly make a differentiation between the terms treatment and conditioning. Conditioning is defined in the international community as an operation producing a waste package suitable for

handling, enclosure of the wate in containers or overpacking. Treatment is defined as operations intended to improve the safety and/or economy by changing the characteristics of the waste through volume reduction, removal of radionuclides and change in composition. US terminology covering both conditioning and treatment is generally referred to treatment or processing.

As noted above, GTCC waste is a form of low-level waste containing long- and short-lived radionuclides with properties dictating a more robust disposal strategy than other classes of LLW. The decommissioning of nuclear power reactors generates GTCC waste and it is noteworthy that typical radionuclides associated with GTCC waste include C-14. LLW may be treated to remove free liquids, stabilise or destroy other hazardous components contained in the waste and/or reduce the final disposal volume through compaction. Based upon information in the second national report for the Joint Convention (October 2005), there are three active licensed commercial LLW disposal sites in the US (GTS-Duratek/Chem-Nuclear, Barnwell, South Carolina; DOE Hanford Site, Richland, Washisngton; Envirocare of Utah, Clive, Utah), however, none can accept GTCC LLW. These are near-surface disposal sites.

3 Policy Objectives of Decommissioning

3.1 NRC Decommissioning Approach

NRC regulations assign responsibility for decommissioning licensed and unlicensed facilities to the licensee or other responsible parties. NRC regulations specify the requirements for a power reactor licensee to provide funds for decommissioning. NRC evaluates the licensee's proposed decommissioning plan, including the licensee's justification for using a particular remediation methodology to determine if it is appropriate. The decommissioning process consists of a series of integrated activities, beginning with the facility in transition from 'active' to 'decommissioning' status and concluding with the termination of the license and release of the site.

Reactor licensees may choose one of three methods for decommissioning their plants : DECON, SAFSTOR or ENTOMB.

- DECON (immediate dismantlement), soon after the nuclear facility closes, equipment, structures and portions of the facility containing radioactive contaminants are removed or decontaminated to a level that permits release of the property and termination of the NRC license.
- SAFSTOR, a nuclear facility is maintained and monitored in a condition that allows the radioactivity to decay, and is later dismantled.
- ENTOMB, radiactive contaminants are encased in a structurally sound material such as concrete and appropriately maintained and monitored until the radiactivity decays to a level permitting release of the property.

Current regulations require decommissioning to be completed within 60 years. ENTOMB is not considered a viable option for reactor decommissioning because some long-lived radionuclides will not have decayed to acceptable levels within this 60 year period. DECON and SAFSTOR options would require access to appropriate LLW, GTCC waste and HLW disposal facilities.

3.2 DOE Decommissioning Approach

The DOE management approach[3] to disposing of excess DOE facilites covers transition, deactivation, surveillance and maintenance and decommissioning.

- Transition – occurs between operations and disposition in a facility lifecycle, beginning once the facility has been declared or forecast to be excess to current and future needs. This phase includes placing the facility in stable and known conditions, identifying hazards and characterising the facility conditions, eliminating or mitigating hazards and conducting stabilisation and transferring responsibilities (programmatic and financial) from the operating to the disposition programme. Materials requiring special handling (including nuclear materials) should be removed at shutdown where possible. During transition, the fate of the facility is determined (re-use through to decommissioning).

- Deactivation – activities during this period include surveillance and maintenance, termination of unneeded facilities, disposal of remaining hazardous chemicals, isolation of equipment and removal of valuable excess equipment. Appropriate characterisation and documentation should be conducted for remaining contamination and waste.
- Post-deactivation, surveillance and maintenance – the facility is in safe storage mode with ongoing low levels of surveillance and maintenance. The facility is generally unoccupied and locked up except for periodic inspections. Depending upon the duration of this phase, refurbishment and repair may be necessary.
- Decommissioning – (and ultimate disposition) will be scheduled in accordance with an overall national priority based on resources.

4 Decommisioning of Nuclear Reactors - Current Situation

According to the DOE 2003 report on nuclear reactors in the US, there are just two operable civilian plant containing graphite, the University of Florida Test Reactor (UFTR) and the General Electric Nuclear Test Reactor (NTR) in Pleasanton, California. Details of these and all shutdown nuclear reactors containing graphite are provided in Table 1. Table 1 has been compiled from information provided on the US DOE website and in particular from the document entitled 'Nuclear Reactors Built, Being Built, or Planned : 2003 In the United States', DOE/NE-0118, December 2003. The status of each reactor (where known) is included, aligning with the categories described in Section 3.

5 Graphite in US Nuclear Reactors

The first graphite used in experimental piles in the early 1940s was an electrode grade graphite AGX produced by the National Carbon Company (now Graftech International) containing 1-2 ppm boron. A somewhat purer graphite was subsequently obtained from the Speer Carbon Company to build the CP1/CP2 pile in Chicago in 1942. In their manufacture, these early

Table 1 – Summary of US Nuclear Reactors containing Graphite

Name	Location	Type	Power MW(t)	Period of operation	Status
Civilian Reactors – Power Reactors Shutdown					
Fort St Vrain	Platteville, Colorado	High Temperature Gas Reactor (HTGR)	842	1974-1989	DECON completed
Peach Bottom Unit 1	Peach Bottom, Pennsylvania	High Temperature Gas Reactor (HTGR)	115	1966-1974	SAFSTOR
Civilian Reactors – Experimental Power Reactors Shutdown					
Molten Salt Reactor Experiment	ORNL, Tennessee	Single region graphite moderated (MSRE)	8	1965-1969	(DOE)
Sodium Reactor Experiment	SSFL, California	Sodium graphite (SRE)	20	1957-1964	Deactivation announced in 1966 (DOE)
Civilian Reactors – University Reactors Operable					
University of Florida	Gainsville, Florida	Modified Argonaut (UFTR), Graphite/water	0.1	1959-	(NRC)
Civilian Reactors – University Reactors Shutdown					
University of California	Los Angeles, California	Educator, Graphite/water	0.1	1960-1984	License terminated in 1993 (NRC)
Iowa State University	Ames, Iowa	Argonaut (UTR-10), Graphite/water	0.01	1959-1998	DECON (NRC)
North Carolina State University	Raleigh, North Carolina	Graphite/water	0.01	1960-1973	License terminated in 1983 (NRC)
Virginia Polytechnic Institute	Blacksburg, Virginia	Graphite/water (UTR-10)	0.1	1959-1984	License terminated in 1988 (NRC)
University of Washington	Seattle, Washington	Argonaut, Graphite/water	0.1	1961-1988	DECON (NRC)
Civilian Reactors – Research and Test Reactors Operable					
General Electric Nuclear Test Reactor	Pleasanton, California	Light Water Reactor (LWR), graphite	0.1	1957-	(NRC)
Civilian Reactors – Research and Test Reactors Shutdown					
American Standard Inc.		Graphite/water (UTR-1)	-	1958-1960	Shipped abroad for exhibition purposes (NRC)
Argonne Low Power Research Reactor	ANL, Chicago, Illinois	Juggernaut, Graphite/water	0.25	1962-1970	(DOE)
Argonne Nuclear Assembly for University Training	ANL, Chicago, Illinois	Argonaut (CP-11), Graphite, water	0.01	1957-1972	(DOE)

Table 1 (cont) – Summary of US Nuclear Reactors containing Graphite

Name	Location	Type	Power MW(t)	Period of operation	Status
Civilian Reactors – Research and Test Reactors Shutdown					
Brookhaven Graphite Research Reactor	BNL, New York State	Air-cooled, graphite moderated (BGRR)	20	1950-1968	Decommissioning underway (DOE)
Chicago Pile 1, rebuilt as CP-2	ANL, Illinois	Graphite (CP-2)	-	1942-1954	Decommissioned and buried on site (DOE)
High Temperature Lattice Reactor	PNNL, Hanford, Washington	Graphite moderated (HTTR)	0.002	1967-1971	(DOE)
Oak Ridge Graphite Reactor	ORNL, Tennessee	Graphite (ORG)	3.5	1943-1963	(DOE)
Physical Constants Test Reactor	PNNL, Hanford, Washington	Graphite (PCTR)	0	1955-1972	(DOE)
Thermal Test Reactor No. 2	PNNL, Hanford, Washington	Graphite (TTR-2)	0	1955-1972	(DOE)
Transient Reactor Test Facility	ANL, Illinois	Graphite (TREAT)	0.12	1959-1994	(DOE)
UTR Test Reactor	-	Graphite/water	-	1961-1963	(NRC)
Production Reactors – Materials Production Shutdown					
B Reactor	Hanford Site, Washington	Graphite	250	1944-1968	(DOE)
C Reactor	Hanford Site, Washington	Graphite	650	1952-1969	(DOE)
D Reactor	Hanford Site, Washington	Graphite	250	1944-1967	(DOE)
DR Reactor	Hanford Site, Washington	Graphite	250	1950-1964	(DOE)
F Reactor	Hanford Site, Washington	Graphite	250	1945-1965	(DOE)
H Reactor	Hanford Site, Washington	Graphite	400	1949-1965	(DOE)
KE Reactor	Hanford Site, Washington	Graphite	1850	1955-1971	(DOE)
KW Reactor	Hanford Site, Washington	Graphite	1850	1955-1970	(DOE)
N Reactor	Hanford Site, Washington	Graphite	4000	1964-1986	(DOE)
Production Reactors – Process Development Shutdown					
Hanford 305 Test Reactor	Hanford Site, Washington	Graphite (HTR)		1944-1976	(DOE)
SR 305-M Test Pile	Hanford Site, Washington	Graphite (Test Pile)		1953-1983	Test pile has been dismantled (DOE)
Standard Pile		Graphite (SP/SE)		1953-1979	(DOE)

graphites were heat-treated to temperatures of 2500-2600°C, a temperature at which much of the impurity originating from the raw materials remained. The first true nuclear graphites were manufactured from more appropriate (purer) raw materials and graphitised at temperatures in excess of 2800°C. Grade AGOT nuclear graphite was employed in the Oak Ridge X-10 reactor (ORG). This graphite was formed and baked to 900°C at the Clarksburg, West Virginia, plant of the National Carbon Company and graphitised to 2800°C at the Morganton, North Carolina, plant. The coke used in the manufacture of AGOT graphite was a highly crystalline and anisotropic material derived from thermal tars as a by-product of the Kendall Oil Refinery, Pennsylvania. It was low in metallic impurities, boron and sulphur (alleviating the need to add iron for its removal). Graphite bars were formed by extrusion.
The X-10 core at Oak Ridge consisted of graphite bars stacked all in a single direction to form a block measuring 24ft (~7.6 m) on each side. The block was pierced by horizontal diamond-shaped channels in which rows of cylindrical uranium slugs formed long rods. Cooling air circulated through the channels.

The Hanford reactors that were to follow were designed with cores in which graphite bricks were stacked in alternating rows transverse to each other. The most complete account of nuclear graphite development for the Hanford reactors is provided by Morgan of Pacific Northwest Laboratory[4]. AGOT graphite was used for the first B, D and F Hanford reactors. The primary grade of graphite for what was termed the green zone was KC (Kendall coke with a semi-needle morphology, Chicago pitch), an extruded material graphitised at 2800°C. It should be noted that both for this first generation of cores and subsequent generations, lower purity graphites were used in the surrounding white zone and reflector regions. The solidly packed cubes of graphite were composed of crossed four by four inch bars through which ran water-cooled aluminium tubes containing slugs of uranium metal. The Hanford DR reactor was the last in the series to use the AGOT graphite based upon Pennsylvania crude. Crude from Texas and Louisiana provided an alternative source of highly anisotropic low sulphur material. However, this had a higher boron level requiring gas purification. A purification process involving freon could reduce boron levels to less than 0.1 ppm. The grade designation for the primary graphite for DR reactor was KCF. A scaled-up process was subsequently

developed by Great Lakes Carbon Company and National Carbon to produce grade CSF graphite, a freon-purified AGOT produced from Louisiana Cleves coke which was used for the H Hanford reactor. In this material, the petroleum coke of conventional morphology was graphitised again at 2800°C. The C Hanford reactor that followed used a new National Carbon graphite GBF (or CSGBF). This material was again produced from Cleves coke but it was 'graphitised' at 2450°C (the lowest temperature at which freon purification could be effected) in an unsuccessful attempt to reduce anisotropy and irradiation-induced dimensional change. In fact, this material was not dissimilar in performance to AGOT. All six reactors measured 28 feet (~8.5 m) from front to rear, 36 feet (~11.0 m) from side to side and 36 feet (~11.0 m) from top to bottom.

The KE and KW Hanford reactors were the last in a series of eight 'single pass' graphite moderated reactors on the Hanford site. The graphite remained essentially the same as that used on the C reactor, GBF grade (or TSGBF), but from a new Texas Lockport coke source with a semi-isotropic morphology. The K reactors measured 33 feet (~10.1 m) from front to rear, 40 feet (~12.2 m) from side to side and 40 feet (~12.2 m) from top to bottom.
The last reactor to be built on the Hanford site was the N reactor which used a new finer grain TSX graphite from Union Carbide. This was a highly anisotropic needle coke material graphitised at 3000°C. This fourth generation reactor built at Hanford was designed to generate electricity as well as producing special materials. The structure measured ~39.5 feet (~12.0 m) from front to rear, 33 feet (~10.1 m) from side to side and ~33.5 feet (~10.2 m) from top to bottom, comprising approximately 1630 tonnes of graphite.

Based upon this reported mass of graphite for the N reactor, the total mass of graphite for the nine reactors on the Hanford site will amount to approximately 14,000 tonnes.
The Brookhaven Graphite Research Reactor (BGRR) was contemporary with the DR Hanford reactor. This was a 25ft (~7.6 m) graphite cube built in two halves separated by a vertical gap. More than 68,000 graphite pieces were used to construct the pile (Figure 1). The pile comprised 75 layers of 4 inch (~100 mm) square graphite blocks of varying length. Five grades of graphite totalling 653 te with varying levels of impurity were used in the structure.

Peach Bottom Unit 1 was an experimental helium-cooled graphite moderated reactor similar in many respects to the Dragon experimental HTR in the UK. The fuel for the Peach Bottom reactor consisted of a uranium-thorium dicarbide kernel, overcoated with pyrolytic carbon and silicon carbide which were dispersed in carbon compacts, and encased in graphite sleeves. The effective height and diameter of the core were 2.3 m and 2.8 m respectively with a ssurrounding graphite reflector of thickness 0.6 m (total mass of graphite approximately 25 tonnes).

The Fort St Vrain nuclear generating station was a high temperature gas cooled reactor (HTGR) of prismatic design employing a highly enriched uranium and thorium fuel cycle, helium cooling and graphite moderation. The reactor core consisted of vertical columns of hexagonal fuel moderator elements and graphite reflector blocks grouped into a cylindrical array within a steel core barrel and supported by a graphite core support structure (Figure 2). The graphite core support floor structure and the large side reflector blocks just inside the core barrel were permanent reactor components. The hexagonal columns inboard were removable by a fuel transfer machine. The fueled core had the approximate shape of a right vertical cylinder with an equivalent diameter of 5.94 m and a vertical height of 4.75 m. A graphite top and bottom reflector, 1.01 m and 1.19m in height respectively, together with a side reflector of mean thickness of 1.19 m surrounded the fuel moderator elements. The entire core weight was ~612 te. As originally constructed, the permanent reflector was grade HLM from Great Lakes Carbon. Core support posts were manufactured from Stackpole grade 2020 (isomolded). Fuel blocks and replaceable reflector were manufactured from H-327 (Great Lakes). In a joint effort between Great Lakes and General Atomics (with assistance from ORNL), a new isotropic petroleum coke graphite grade (H-451) was later developed and used for replacements so that eventually the Fort St Vrain whole core was H-451.

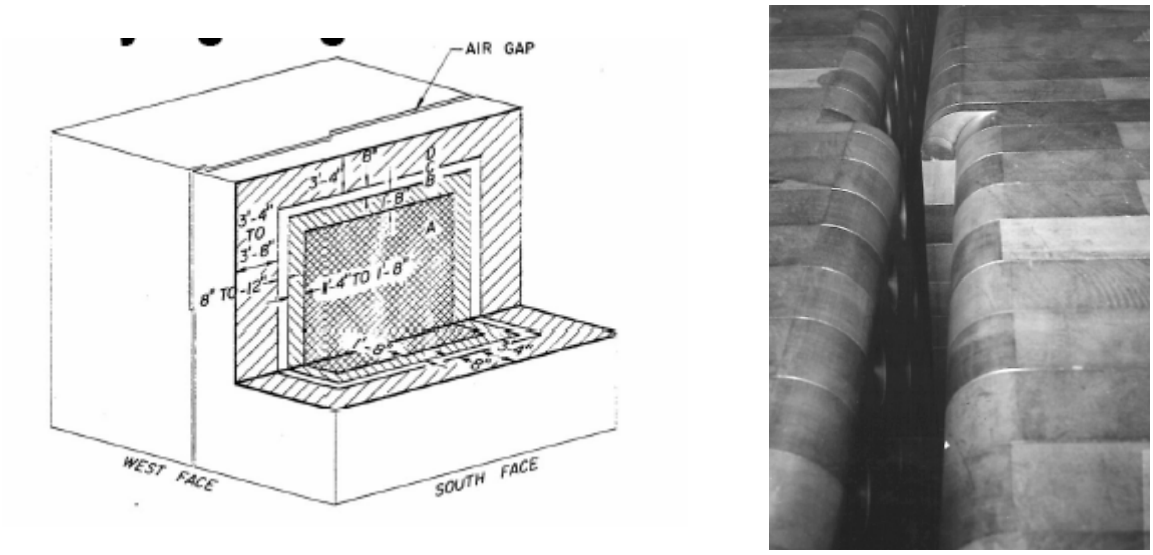

Figure 1 : Brookhaven Graphite Research Reactor, cut-out section through core and view of graphite blocks (from Brookhaven Graphite Research Reactor (BGRR) Decomissioning Project presentation by T Kneitel, October 2007).

6 Decommissioning Strategy

The current decommissioning status of US nuclear reactors containing graphite is summarised in Table 1.

Those commercial plant under the jurisdiction of the NRC are the HTGRs at Peach Bottom and Fort St Vrain. Peach Bottom Unit 1, which closed in 1974, currently has a SAFSTOR decommissioning status. Under this status, the nuclear facility is being maintained and monitored in a condition that allows the radioactivity to decay with the intention of dismantling at some unspecified time in the future. The Fort St Vrain plant, which closed in 1989, has been dismantled and has a DECON decommissioning status. Under this status, equipment, structures and portions of the facility containing radioactive contaminants have been removed and the site has been restored to brown-field. A disposal route was already in place for the graphite fuel blocks, these being designed to be removed and replaced as part of routine operation with dedicated facilities available to receive them. Reflector blocks were removed,

along with other pressure vessel internals, after cutting access through the concrete top head section and flooding the pressure vessel with water to minimise radiation dose to the operators[5]. A rotating service platform was inserted in order to transfer the graphite to a transfer basket which was raised into a 'shield bell' and the contents transferred to shipment flasks for disposal as low-level waste.

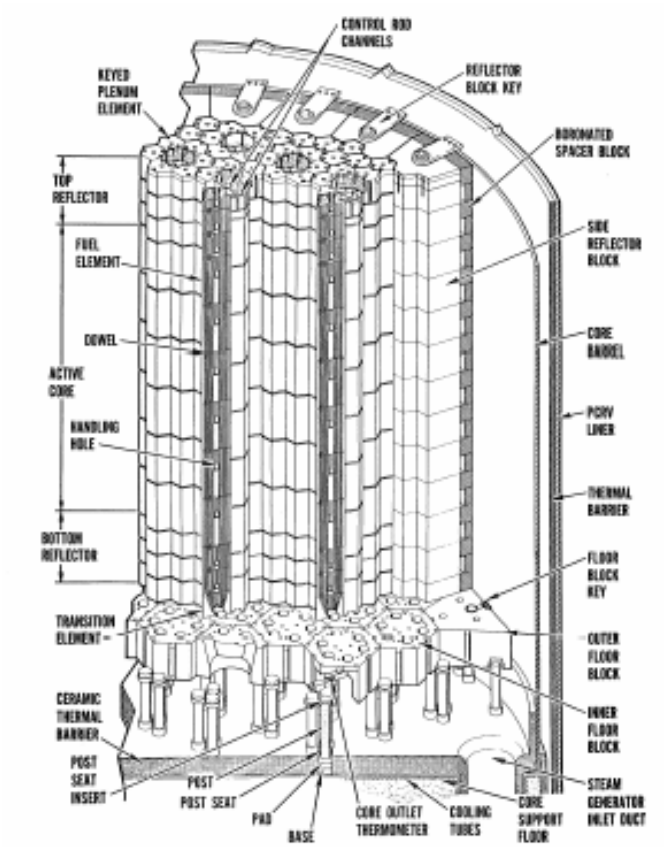

Figure 2: Fort St Vrain reactor core arrangement (from 'Design features of the core and support structures for the Fort St Vrain nuclear generating station', D A Nehrig, A J Neylan and E O Winkler, Graaphite Structures for Nuclear Reactors, The Institution of Mechanical Engineers, London, March 1972)

The most significant plant containing i-graphite under the jurisdiction of the DOE are the Hanford reactors and the Brookhaven Graphite Research Reactor. The Hanford reactors are in the deactivation/post-deactivation phase. No decision has yet been reached on the ultimate

disposal of the N reactor. There are proposals for the other eight reactors, in which the moderators will be transported as single units encased in the reactor shielding from their existing sites near the Columbia River to higher ground. They will then be buried in a massive pit, prepared especially for their disposal near the centre of the Hanford site. The Brookhaven Graphite Research Reactor is currently undergoing decommissioning. It is intended that the graphite blocks will be lifted out of position using a remote handling device (Brokk) and placed in soft-sided bags which in turn will be sealed in steel boxes. This phase of the decommissioning has not yet commenced. The steel waste boxes will be disposed of at the DOE LLW site in Nevada.

As shown in Table 1, the Chicago Pile 1 subsequently rebuilt as CP-2 was decommissioned and buried on site, under the jurisdiction of the DOE. The SR 305-M Test Pile at Hanford has also been dismantled and the graphite stored on a DOE site.

7 References

[1] Nuclear Reactors Built, Being Built, or Planned: 2003 In the United States, DOE/NE-0118, December 2003.

[2] United States of America Second National Report for the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management, DOE/EM-0654, Rev 1, October 2005.

[3] DOE Order 430.1A, Life Cycle Asset Management.

[4] Nuclear graphite development, operational problems, and resolution of these problems at the Hanford production reactors, W C Morgan, IAEA-TECDOC-901 Graphite moderator lifecycle behaviour, August 1996.

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