U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

Technical Letter Report

TLR-RES/DE/REB-2025-20

Survey of Nuclear Graphite Waste Regulations

Date:

January 2026

Prepared under CONTRACT NO. 31310024P0040 by:

Anthony J. Wickham
Werner von Lensa
Makuteswara Srinivasan
Arthur Desrosiers
NUMARK Associates, Inc.

NRC Project Manager:

Joseph Bass
Reactor Engineer
Reactor Engineering Branch

Division of Engineering
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555–0001

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the U.S. Government. Neither the U.S. Government nor any agency thereof, nor any employee, makes any warranty, expressed or implied, or assumes any legal liability or responsibility for any third party's use, or the results of such use, of any information, apparatus, product, or process disclosed in this publication, or represents that its use by such third party complies with applicable law.

This report does not contain or imply legally binding requirements. Nor does this report establish or modify any regulatory guidance or positions of the U.S. Nuclear Regulatory Commission and is not binding on the Commission.

DISCLAIMER

Neither NUMARK, nor any of their employees or contractors, makes any warranty, expressed or implied, or assumes any legal liability or responsibility for any third party's use, or the results of such use, or any information, apparatus, product or process disclosed in this publication, or represents that its use by such third party complies with applicable law.

COVER PHOTO: Refueling floor at Fort Saint Vrain HTGR, 1972. Bruce McAllister - U.S. National Archives and Records Administration. Public Domain.

ACKNOWLEDGEMENTS

This research was sponsored by the U.S. Nuclear Regulatory Commission under a contract to NUMARK Associates, Inc. Dr. Joseph Bass of the Office of Nuclear Regulatory Research was the Technical Monitor. We acknowledge his and Dr. Raj Iyengar's technical direction in conducting this research. We also acknowledge other NRC staff's reviews and comments.

Contents

LIST OF FIGURES

LIST OF TABLES

ABBREVIATIONS AND ACRONYMS

Abbreviation	Meaning or Expansion
AGR	(British) Advanced Gas Reactor
AHTR	Advanced High-Temperature Reactor
ALI	Annual Limit on Intake
ANDRA	Agence Nationale pour la Gestion des Déchets Radioactifs (France)
ARI	Activation Relevant Impurities
ASN	Autorité de Sûreté Nucléaire (France)
ASTM	ASTM International
AVR	Arbeitsgemeinschaft Versuchsreaktor (Association of Experimental Reactor)
BASE	Federal Office for the Safety of Nuclear Waste Management (Germany)
BfS	and Bundesamt für Strahlenschutz (Germany)
BGRR	Brookhaven Graphite Research Reactor
CARBOWASTE	Treatment and Disposal of Irradiated Graphite and other Carbonaceous Waste
CAST	CArbon-14 Source Term
CFR	Code of Federal Regulations
CORDIS	Community Research and Development Information Service
DDEP	Decay Data Evaluation Project
DOE	U.S. Department of Energy
DOT	U.S. Department of Transportation
EDF	Électricité de France
EIS	Environmental Impact Statement
ENRESA	Empresa Nacional de Residuos Radiactivos (Spain)
EU	European Union
FFE	Fissile/Fissionable Element
FHR	Fluoride Salt-Cooled High-Temperature Reactor

GCC	Graphite Core Component
GRAPA	Irradiated Graphite Processing Approaches (IAEA Collaborative Study)
GSG	(IAEA) General Safety Guide
GTCC	Greater than Class-C
HEPA	High Efficiency Particulate Air
HLW	High-Level Waste
HTGR	High-Temperature Gas-Cooled Reactor
HTR	High-Temperature Reactor
HTR-PM	High-Temperature Gas-cooled Reactor Pebble-modules Demonstration Project
HTTR	High-Temperature Test Reactor (Japan)
IAEA	International Atomic Energy Agency
ILW	Intermediate-Level Waste
IP	Industrial Package
JAEA	Japanese Atomic Energy Agency
KAERI	Korea Atomic Energy Research Institute
LLW	Low-Level Waste
LSA	Low Specific Activity
MAGNOX	Magnesium Non-oxidizing
MCRI	Metallic Corrosion Relevant Impurities
MIE	Minimum Ignition Energy
MSR	Molten-Salt Reactor
NACOK	NAturzug im COre mit Korrosion (Natural convection in the core with corrosion) test facility at FZ-Juelich, Germany
NAM	Neutron Absorbing Impurities (NAM)
NEA	Nuclear Energy Agency
NNSS	Nevada National Security Site
NPP	Nuclear Power Plant
NRA	Nuclear Regulation Authority (Japan)

NRC	U.S. Nuclear Regulatory Commission
NUREG	Nuclear Regulatory Guidance
OECD	Organization for Economic Cooperation and Development
ONR	Office for Nuclear Regulation (U.K.)
OPC	Oxidation Promoting Catalysts
PNNL	Pacific Northwest National Laboratory
QA	Quality Assurance
RBMK	Russian: Реактор большой мощности канальный, РБМК; reaktor bolshoy moshchnosti kanalnyy, "high-power channel-type reactor"
RCRA	Resource Conservation and Recovery Act
SRP	Standard Review Plan
SSR	(IAEA) Specific Safety Requirements
TECDOC	IAEA Technical Document Series
TEDE	Total Effective Dose Equivalent (10 CFR § 20.1003)
TLR	Technical Letter Report
TRL	Technical Readiness Level
TRU	Transuranic
U.K.	United Kingdom
U.S.	United States of America
VATESI	State Nuclear Power Safety Inspectorate (Lithuania)
VHTR	Very High-Temperature Reactor
VLLW	Very Low-Level Waste
WAC	Waste Acceptance Criteria
WIPP	Waste Isolation Pilot Plant

EXECUTIVE SUMMARY

This report provides a review of managing irradiated graphite and carbonaceous waste, with a focus on regulatory, technical, and operational dimensions in the U.S. and internationally. The growing inventory of irradiated graphite—estimated at over 250,000 metric tons globally—requires coordinated efforts to ensure safe, cost-effective, and environmentally sound waste management.

Irradiated graphite and carbonaceous waste are characterized by high concentrations of tritium and of long-lived radionuclides such as ^{14}C and ^{36}Cl, which complicate classification.

The Nuclear Regulatory Commission's (NRC) performance-based regulations, including Title 10 of the Code of Federal Regulation (CFR) Parts 20, 50, 61, 71, and 72, are generally technology-neutral. They allow for licensee flexibility but require demonstration of safety for any proposed treatment, conditioning, or disposal approach.

In the U.S., for advanced helium or molten-salt cooled high-temperature reactors with tri-structural isotropic (TRISO) fuel, much of this material would exceeds Class C limits under NRC's 10 CFR Part 61 and would be designated greater-than-class-C (GTCC) waste, for which no licensed commercial disposal facility currently exists. As a result, onsite or offsite interim storage would be the default strategy.

Various volume reduction and treatment methods could be utilized, including oxidation, thermal treatment, other chemical treatments (including electrochemical treatments), and/or microbial degradation. A treatment facility that handles irradiated graphite and/or fuel elements would require licensing according to nuclear regulations for both normal operation and accident conditions. Regulatory approval for any of these treatments would require proof of radionuclide control, waste form stability, and public/environmental protection.

International practices vary widely. The U.K. and Russia favored long-term interim storage although the U.K., via Nuclear Restoration Services, is looking to speed up dismantling of the Magnox and Advanced Gas-cooled Reactors (AGR) reactors in order to reclaim the licensed sites for reuse with small modular reactors (SMRs). France and Germany package graphite for eventual geological disposal. Spain and South Korea apply engineered near-surface storage strategies, and China is exploring volume reduction and decontamination techniques. These practices offer valuable insights: early waste characterization, modular container design, and flexible classification systems can help improve graphite waste management.

Key regulatory and technical considerations include the development of pilot-scale or commercial treatment processes and managing ^{14}C and other radioactive emissions from thermal or chemical treatments, ensuring long-term stability of storage systems, qualifying waste forms under performance assessment models, and coordinating cross-agency jurisdiction between the NRC, Department of Energy (DOE), Department of Transportation (DOT), and Environmental Protection Agency (EPA). When assessing environmental impact, long-term scenarios involving ^{14}C migration are an important consideration. While current graphite disposal practices are safe and conservative, there could be benefits from increased knowledge and understanding of ^{14}C reduction/stabilization, advanced leaching, robotic dismantling of graphite cores, and improved performance modeling.

This report concludes that NRC regulations are largely technology-neutral and sufficiently flexible to

accommodate new graphite waste technologies, provided supporting consolidated process and safety data are available.

SURVEY OF GRAPHITE WASTE REGULATIONS

1. INTRODUCTION

This technical letter report (TLR) contains research resulting from a review of the current regulatory framework relevant to graphite waste. Regulations and guidance on transportation (including 10 CFR 71 and NUREG 2216) and storage (including 10 CFR 72 and NUREG 2215) are considered in this review, in addition to 10 CFR Part 20, 10 CFR Part 61 and others. Where applicable and available, International Atomic Energy Agency (IAEA) standards and the regulatory requirements of the United Kingdom (U.K.), Germany, Spain, France and China were also examined. Methodologies implemented in the U.K. for addressing the graphite waste from their graphite-cored reactors, as well as domestic knowledge from Peach Bottom and Fort Saint Vrain were reviewed. This review augments the information released in previous TLRs (such as ML20237F397 and ML25350C188).

The Nuclear Regulatory Commission (NRC) waste transportation and storage regulations and standards were reviewed to assess their technology neutrality, with respect to the sources which produced the waste. It was concluded that NRC regulations are largely technology-neutral and sufficiently flexible to accommodate new graphite waste technologies, provided supporting consolidated process and safety data are available.

The report begins with background information on the nature of graphite core components (GCC) in Section 2. In Section 3, the research method covers information on literature review, document analysis, modeling and experimental data collection, and case studies. Section 4 contains detailed discussions on the results of this research and analysis of the information. The major topics covered are the regulatory framework for irradiated graphite low-level waste (LLW) or Greater-Than-Class-C (GTCC), NRC waste classification and disposal requirements, IAEA graphite waste classification, and U.S. Department of Transportation (DOT) regulations. Section 5 contains detailed analysis and evaluation of irradiated graphite material behavior and treatment technologies. As a part of the discussion on integration of regulatory requirements and industry practices, waste classification and disposal paths, considerations for direct disposal versus volume reduction, transportation strategies, technical challenges for NRC and waste management, cask design and storage system evaluation, waste form classification and acceptance criteria, environmental impact assessment, long-term integrity of storage, and cross-agency coordination were explored. Section 6 provides detailed survey and analysis of conditioning and interim storage. In Section 7, key technical and non-technical challenges associated with harmonizing the transportation and storage of irradiated graphite (i-graphite) waste from high-temperature reactors (HTGR) and MSR are covered. Section 8 provides information on volume reduction and separation. Section 9 provides a detailed discussion on Technology Levels and regulatory implications of graphite waste treatment methods. Section 10 provides an assessment of NRC regulations for technical neutrality in managing irradiated graphite waste. Section 11 addresses technical challenges for regulators (NRC) and waste managers. Finally, in Section 12, a summary of this research conclusion is presented.

This report also contains comprehensive information on various topics related to irradiated graphite and its waste management in several Appendices. Appendix A discusses the purity of nuclear grade graphite and neutron activation products resulting in nuclear reactor operation. Appendix B contains information on decay characteristics of some commonly found radioisotopes in irradiated graphite. Appendix C contains a discussion on stored (Wigner) energy and the propensity for oxidation of reactor graphite. In Appendix D, the results of an analytical study of the self-limiting radionuclide decay characteristics in irradiated graphite are discussed with their effects in assuring adequate safety in long-term storage. A potential issue for HTGRs

and MSRs is the graphite dust generated and its associated trapped radionuclides which could deposit on GCC and reactor internals, such as piping. A short discussion of this is provided in Appendix E. Finally, interim storage is discussed in Appendix F.

2. BACKGROUND

Irradiated graphite, graphitic matrix[1] of TRISO fuel and baked carbon constitute a group of irradiated carbonaceous wastes. The waste originates from neutron-activated carbonaceous core components in nuclear reactors, such as moderator, reflector, fuel blocks, graphitic matrix of TRISO fuel and core support columns or thermal insulation layers in gas-cooled reactors. Worldwide, decommissioning activities are estimated to have accumulated in excess of 250,000 tonnes (275,580 U.S. short tons) of irradiated graphite in need of management (Wickham, Von Lensa, Srinivasan, & Desrosiers, 2025). This material is typically categorized as low- to intermediate-level radioactive waste[1] due to its contamination with neutron-activated products and, in some cases, residual fission products. Pertinent to the current research, it is important to examine the purity of as-manufactured graphite, which has evolved over the last few decades to be highly pure. An analysis of the chemical analysis of the various graphites used in past graphite-moderated reactors is detailed in Appendix A.

The radionuclide inventory of irradiated graphite is unusual compared to other wastes: short-lived isotopes like cobalt-60 (^{60}Co)[2] and tritium (Gainey, 1976) dominate the initial activity, whereas long-lived isotopes like carbon-14 (^{14}C) and chlorine-36 (^{36}Cl) govern long-term concerns. Trace quantities of plutonium, and other transuranic (TRU) radionuclides can also be present from defective fuel or from residual uranium content in the nuclear graphite. Each ton of graphite may contain tens of terabecquerels of tritium and significant activity of ^{14}C, underscoring the need for careful control. A simplified overview of the decay characteristics and the estimation protocol is given in Appendix B.

The stored energy due to neutron bombardment, and the combustion and explosion feasibility of reactor-irradiated graphite during any treatment were also examined, as related to such hazards being a part of the analysis needed to ensure safe operations during packaging, transportation, and storage. A brief note on these issues, pertaining to modern HTGRs and MSRs, considering their fuel, coolant, and operating temperatures, finds them to be of only minor concern: an evaluation of the actual contribution of graphite to Windscale and Chernobyl reactor accidents as well as air-ingress scenarios in future HTGRs or high-temperature treatment processes, is provided in Appendix C.

A robust framework of regulations and guidelines governs irradiated graphite LLW management. The NRC classifies commercial LLW as Class A, B, C, or GTCC based on radionuclide concentrations (10 CFR 61.55). Graphite waste often contains long-lived radioisotopes (like ^{14}C) at levels that exceed Class C limits for near-surface disposal, effectively making it GTCC waste in the U.S. classification system. Such waste is not generally acceptable at shallow land disposal facilities and may need to be disposed of in a geologic repository unless alternate methods are approved or treatment processes allow a reclassification into Class A to Class C. The U.S. DOE, responsible for government-owned graphite waste and GTCC-like waste, follows DOE Order 435.1 and has conducted a GTCC Environmental Impact Statement (EIS) to evaluate disposal options such as deep geological disposal, boreholes, or enhanced near-surface facilities (DOE 2016).

---

[1] The term ‘Graphitic Matrix’ is used in this report when specifically addressing the carbonaceous material which make up the fuel elements and contains the TRISO particles. Spherical fuel elements (pebbles) or cylindrical fuel rods (compacts) inserted into graphite blocks are considered to be fuel elements..

[2] The creation of Cobalt-60 (^{60}Co) in High-Temperature Gas-Cooled Reactors (HTGRs) is primarily associated with the activation of structural materials, particularly those containing cobalt impurities. During reactor operation, neutron irradiation can transmute stable cobalt isotopes into radioactive ^{60}Co, which can then migrate within the reactor coolant circuits.

Internationally, the IAEA provides guidance through technical reports and projects - e.g., GRAPA (Wickham, Steinmetz, O'Sullivan, & Ojovan, 2017; IAEA 2025), CARBOWASTE (von Lensa, et al., 2008) - to assist member states in managing irradiated graphite. Waste classification and acceptance criteria vary by country, from intermediate-level waste (ILW)[3] requiring deep or intermediate-depth disposal in many European nations to strategies of interim safe storage where disposal facilities are not yet available.

We note that NRC has been actively evaluating waste disposal rulemaking (NRC 2024), as detailed on its "Low-Level Radioactive Waste Disposal Rulemaking" webpage and "Greater-Than-Class C and Transuranic Waste" (SECY-24-0045). NRC has also released Draft NUREG-2175, which drafts guidance for conducting technical analysis for 10 CFR Part 71 (NRC 2015). Draft NUREG-2175 is not written relative to the current 10 CFR 61, but is written relative to a proposed 10 CFR 61, as if certain proposed amendments to 10 CFR 61 have been enacted. Within Draft NUREG-2175, Appendix A highlights changes proposed to 10 CFR Part 61 in 2015 that identify new requirements, particularly relating to technical analyses and waste acceptance criteria. Further proposed amendments have also been considered, since 2015. As the proposed amendments have not been enacted, they will not be included in the current report, and the 10 CFR Part 61 that is in effect as of September 1st, 2025, is the version that will be referred to within this report. The Draft NUREG notes that even with the proposed changes to CFR Part 61, the regulation "is performance based, and the technical criteria are written in relatively general terms, which allow applicants to demonstrate how their proposals meet the respective performance objectives for the specific near-surface disposal method selected." Additionally, certain considerations within Draft NUREG-2175 apply under the current 10 CFR Part 61. For example, a disposal facility design should protect an inadvertent intruder at all times during the compliance period after active institutional controls are removed. Draft NUREG-2175 covers general technical analyses considerations such as data adequacy and uncertainty, assessment context system description, scenario development, conceptual model development, numerical model development, and numerical model assessment. Draft NUREG-2175 also covers performance assessment including various phenomena associated with source terms (inventory, chemical environment, waste container, waste form), release (aqueous, gaseous, direct, biota enhanced), radionuclide transport (groundwater, surface water, atmospheric, biotic). Draft NUREG-2175 additionally includes a section on Site Stability analyses, covering disruptive processes and events (natural and anthropogenic), as well as engineered barriers. The Draft NUREG also includes sections on defense-in-depth, waste acceptance, and tables of related NRC guidance documents. Though the document is a Draft NUREG, it includes much factual information as well as useful 'crosswalk' tables to relate to the proposed 10 CFR 61 (as if certain amendments have been enacted) as well as references to existing NRC guidance documents.

The U.S. DOT regulations (49 CFR and aligning with IAEA transport regulations) govern the safe transportation of radioactive graphite, including packaging requirements for Low Specific Activity (LSA) material and Type B shipments for higher-activity contents.

Effective management of irradiated carbonaceous waste LLW requires an interplay of regulatory compliance, engineering controls, and scientific understanding. This report: (1) reviews the current regulations and policies of NRC, IAEA, DOE, and DOT as they pertain to irradiated carbonaceous waste; (2) examines advanced modeling and experimental studies on carbonaceous waste behavior and treatment; (3)

[3] The U.S. Nuclear Regulatory Commission (NRC) does not have a specific classification termed "Intermediate- Level Radioactive Waste." A comparison of NRC's and IAEA's waste classification and categorization is provided in Table 2.

highlights industry best practices in reducing waste volume, transportation, and storage (with relevant case studies); (4) identifies technical challenges for regulators and waste managers, including those faced by NRC staff in evaluating packages and storage solutions; and, (5) identifies future directions and research needs to improve carbonaceous waste management.

This report consolidates information from primary sources, government reports, and peer-reviewed studies, pertaining to regulating irradiated carbonaceous waste management.

3. METHODOLOGY

This technical review used a survey of a wide range of primary literature, including regulatory documents, technical standards, research publications, and industry case studies.

Document Analysis

Key regulations and guidance from NRC (e.g., 10 CFR Part 61 for waste classification, 10 CFR Part 71 for transport, NUREGs), DOE (Order 435.1 and GTCC EIS documentation), IAEA (Technical Document (TECDOC) series and safety standards), and DOT (49 CFR for Class 7 material shipping) were examined to extract requirements specific to irradiated graphite and other carbonaceous waste. Agency positions and policies were cross-referenced to ensure comprehensive coverage of compliance obligations.

The following key NRC reference documents were consulted for this research.

NUREG-0800 – Standard Review Plan: https://www.nrc.gov/reading-rm/doc- collections/nuregs/staff/sr0800/

NUREG-0945 – Disposal of Radioactive Waste on Land: https://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr0945/

NUREG-2157 – Generic EIS for Continued Storage of Spent Nuclear Fuel: https://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr2157/

NUREG-1854 – NRC Review of DOE Waste Determinations: https://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1854/

Regulatory Guide 1.21 – Radioactive Discharges: https://www.nrc.gov/reading-rm/doc-collections/reg-guides/power-reactors/rg/01-021/

Regulatory Guide 1.143 – Design of Waste Systems: https://www.nrc.gov/reading-rm/doc- collections/reg-guides/power-reactors/rg/01-143/

Regulatory Guide 4.15 – QA for Radiological Monitoring: https://www.nrc.gov/reading-rm/doc-collections/reg-guides/environmental-siting/rg/04-015/

SRP 11.4 – Solid Waste Management System: https://www.nrc.gov/docs/ML0635/ML063540293.pdf

SRP 11.5 – Radiological Monitoring Instrumentation: https://www.nrc.gov/docs/ML0635/ML063540295.pdf

SRP 13.3 – Emergency Planning: https://www.nrc.gov/docs/ML0707/ML070740002.pdf

10 CFR Part 61 – Land Disposal Requirements: https://www.nrc.gov/reading-rm/doc- collections/cfr/part061/

10 CFR Part 71 – Land Disposal Requirements: https://www.nrc.gov/reading-rm/doc- collections/cfr/part071

10 CFR Part 20 – Radiation Protection Standards: https://www.nrc.gov/reading-rm/doc-collections/cfr/part020/

Literature Review

Scientific and engineering literature on irradiated graphite was reviewed to gather advanced modeling data and experimental results. This included IAEA-sponsored studies like the CARBOWASTE project on treatment and disposal of irradiated graphite, the European Commission CArbon-14 Source Term (EU CAST) project focusing on ^{14}C release behavior, and peer-reviewed articles on graphite waste characterization.

For example, the behavior of ^{14}C in irradiated graphite has been extensively studied through leaching experiments to simulate geological disposal conditions (Bucur, Ichim, & Florea, 2018). Similarly, material

science investigations into graphite's microstructural changes and Wigner energy retention provided insights into safe handling limits (Gallego & Burchell, 2011, Zhou, 2022).

Other carbonaceous waste, such as irradiated graphitic matrix and baked carbon, are minimally addressed in the related literature, due to the focus on irradiated graphite. The relevance of the non-graphite carbonaceous waste is indicated in (Kim 2024 and von Lensa et al. 2011)

Modeling and Experimental Data Collection

This review emphasizes studies that used neutron activation modeling and material performance simulations to predict irradiated graphite behavior. Neutron transport and activation codes (e.g., ORIGEN and MCNP) are commonly used to calculate radionuclide inventories in reactor graphite based on operating history, neutron flux distributions and impurity content (IAEA, 2006). Contributors to radioactivity (like ^{14}C from ^{13}C or ^{14}N, ^{36}Cl from chlorine impurities, and various activation metals) supports classification of the waste (Cox, Harker, Wickham, & Noy, 2011).

On the experimental side, published leaching tests, oxidation trials, and mechanical property measurements were evaluated. Leaching experiments (e.g., immersing irradiated graphite in aqueous solutions under aerobic and anaerobic conditions) provide data on the release rates of ^{14}C, tritium, and other radioisotopes, which inform safety analyses for disposal facilities (Blackwood, Otlet, & and MacKenzie, 1992).

Laboratory and pilot-scale oxidation and combustion experiments were reviewed to evaluate techniques for volume reduction and ^{14}C extraction. For example, graphite was oxidized under controlled conditions and ^{14}C was trapped as CO_2 for subsequent immobilization (Fuks et al, 2020).

Case Studies

This review examined documented case studies of graphite waste handling. Notable examples include the decommissioning of the Brookhaven Graphite Research Reactor (BGRR) in the U.S., where ~ 700 tons of graphite were removed and packaged for disposal (Brookhaven National Laboratory (BNL), 1997), as well as international cases like the Windscale Piles (U.K.) and reactor dismantlement projects in France, Spain, and Lithuania (Perera, 2024). These case studies provided practical insights into logistical challenges, packaging solutions, and stakeholder considerations in real projects.

By combining regulatory analysis with scientific data from modeling and experiments, this report's methodology ensures a well-rounded and information-inclusive perspective. The knowledge gathered was synthesized to identify current best practices and challenges. All sources of data are cited for transparency and verification. This approach enables a combined examination of irradiated graphite waste management from both a policy/regulatory standpoint and a technical performance standpoint.

4. RESULTS AND TECHNICAL ANALYSES

U.S. REGULATORY FRAMEWORK FOR IRRADIATED GRAPHITE

There exist variations in the waste classification schemes between NRC and IAEA. As HTGRs and MSRs advance, harmonizing graphite waste disposal strategies is desirable for regulatory consistency and long-term waste management planning. Key differences and implications for NRC policy are evaluated below.

The NRC's regulatory process for establishing a new facility to store GTCC reactor-irradiated graphite waste follows the procedures shown in Figure 1. A similar process would be followed to add a new waste stream to an existing GTCC storage facility. In either case, any waste that is within LLW limits could be disposed of in an existing licensed facility if the facility's acceptance criteria were met.

Table 1 summarizes key NRC regulatory requirements, guidance documents, and review criteria applicable to the management, storage, transportation, and disposal of irradiated graphite waste. This flow chart would specifically be for a new storage facility license. It is intended to support the preparation of compliant submissions for licensing, storage, or disposal under NRC authority.

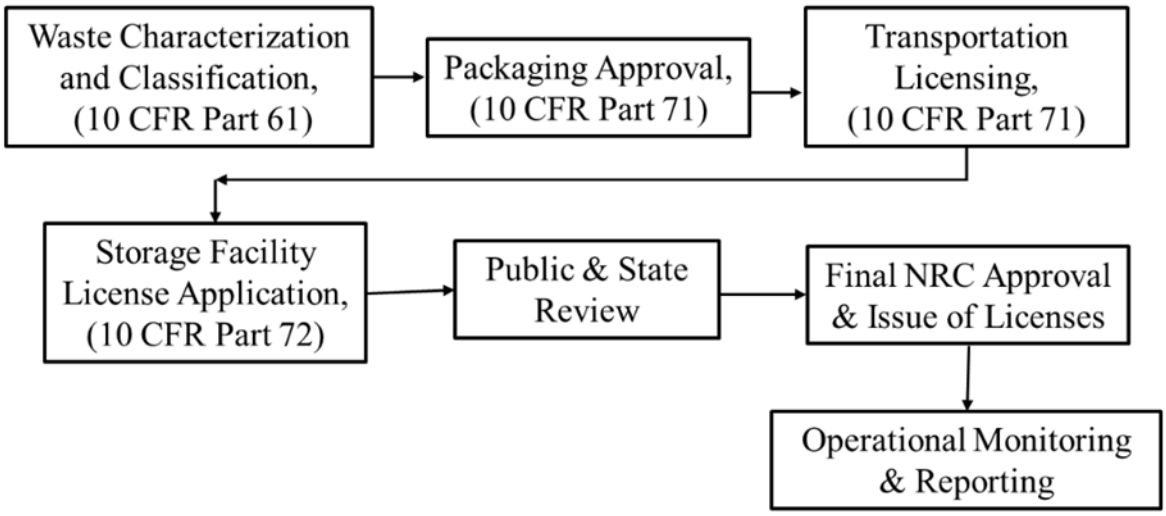

Figure 1. Flow Chart for NRC Regulatory Process for Graphite Waste Transport and Storage

Table 1. NRC Regulations and Guidance Documents Related to Irradiated Graphite Storage and Disposal

Regulatory/Guidance Document	Requirement/Guidance Summary	Applicability to Irradiated Graphite
10 CFR Part 20	Radiation protection standards	Applies to handling, worker exposure, environmental releases, and monitoring during storage and processing
10 CFR Part 61	Land disposal requirements	Applies if graphite is classified as Class B/C or GTCC waste, requiring disposal performance assessment
NUREG-0800 (SRP 11.4)	Solid waste management system	Covers design and operation of graphite storage and conditioning systems at reactor sites or waste facilities
Regulatory Guide 1.143	Design guidance for waste management systems	Provides structural, seismic, and environmental design criteria for waste storage containers and facilities
NUREG-0945	EIS for land disposal	Relevant if disposal of irradiated graphite requires a new or revised license application for a disposal site
Regulatory Guide 4.15	Quality assurance for radiological monitoring	Covers monitoring programs for vented containers, effluent sampling, and long-term environmental surveillance
NUREG-2157	Generic EIS for continued storage of spent nuclear fuel	Provides precedents for environmental and safety analysis of long-term dry-storage systems (relevant to graphite packages)
NUREG-1854	Guidance on DOE waste determinations	Applies if DOE graphite waste streams are subject to waste determination under DOE Order 435.1

Additional Potential Practices

- Conduct early characterization to establish an accurate radionuclide inventory (including ^{14}C and ^{36}Cl).
- Perform radiolysis gas generation modeling to support container design and venting requirements.
- Evaluate the need to discharge of accumulated water (see Appendix) or and apply inert gas atmospheres.
- Ensure container design meets seismic, fire, and corrosion resistance requirements per Reg

Guide 1.143.

- Establish long-term gas sampling and container inspection programs for interim storage.
- Document all characterization, modeling, and operational history in the final waste package record.
- Align waste form performance assessment (for Part 61 compliance) with actual site-specific geochemistry and hydrogeology.

NRC Waste Classification and Requirements

This section provides a regulatory overview of how irradiated graphite is classified under NRC rules and the applicable requirements under 10 CFR Parts 20, 50, 61, and 71. The regulatory framework overview is depicted in Table 2.

Table 2. Regulatory Framework Overview for Irradiated Graphite

Regulation	Purpose	Key Applicability to Irradiated Graphite
10 CFR Part 20	Radiation protection standards	Establishes effluent and occupational exposure limits for radionuclides in air and water. Used to ensure protection of workers and the public.
10 CFR Part 50	Domestic licensing & decommissioning	Governs the decommissioning process for nuclear facilities. Requires demonstration that residual radioactivity (e.g., from irradiated graphite) does not exceed dose limits (≤ 25 mrem/year TEDE) to support license termination.
10 CFR Part 61	Land disposal of radioactive waste	Classifies LLW into Classes A, B, and C; defines GTCC for wastes exceeding Class C limits. Often applicable to irradiated graphite due to long-lived radionuclides.
10 CFR Part 71	Packaging and transportation	Specifies requirements for transportation of radioactive materials including irradiated graphite. Ensures containment, shielding, and accident resistance.
10 CFR Part 72	Licensing requirements for independent storage of waste	Storage of irradiated graphite is regulated under 10 CFR Part 72, which governs the independent storage of GTCC.

Under NRC regulations (10 CFR Part 61), irradiated graphite from commercial reactors is often place it in the highest classification because of its long-lived radionuclides. ^{14}C, with a half-life of 5,730 years, is a key isotope: NRC sets a concentration limit of eight Curies per cubic meter for ^{14}C in LLW to qualify for near-surface disposal (Class C).

Based on IAEA TECDOC-2072 and NUREG/CR-6944, Table 3 summarizes the regulatory thresholds and GTCC designation potential for major radionuclides found in irradiated graphite.

Per 10 CFR 61.55, waste exceeding the Class C limits is classified as GTCC. NRC staff and DOE's GTCC EIS (DOE/EIS-0375D) confirm that irradiated graphite from HTGRs, such as fuel blocks and reflectors, frequently exceed Class C thresholds due to ^{14}C and ^{36}Cl, making it GTCC. In Table 3, the 10 CFR 61.55(a) and 10 CFR 20 limits for radionuclides of concern in irradiated graphite are provided. Long-lived radionuclides (e.g. ^{14}C, ^{36}Cl, ^{63}Ni) are in Table 1 of § 61.55(a). Short-lived radionuclides (e.g. ^{90}Sr,^{137}Cs) are

in Table 2 of § 61.55(a). For nuclides not listed, the regulation defers to practical considerations instead of numeric limits. In 10 CFR 61.55, practical consideration "such as the effects of external radiation and internal heat generation on transportation, handling, and disposal" become important in developing concentration limits.

Table 3. Radionuclides of Concern in Irradiated Graphite (IAEA TECDOC-2072 and NUREG/CR-6944)

Radionuclide	**Half-Life (Years)**	**10 CFR 61.55(a) Class C Limit (Ci/m^3)**	**Expected in Graphite?**	**GTCC, if Exceeded?**	10 CFR 20, Appendix B		
					Effluent Limit, (µCi/mL)	**ALI (Oral), (mCi)**	**ALI (Inhalation) (mCi)**
^{14}C	5730	8	Yes	Yes	Water: $3x10^{-6}$ Air: $2x10^{-8}$	200	100
^{36}Cl	$30x10^{3}$	3	Yes	Yes	Water: $6x10^{-6}$ Air: $1x10^{-7}$	400	200
^{3}H	12.3	Operational considerations apply	Yes	Yes	Water: $1x10^{-3}$; Air: 10^{-7}	80,000	40,000
^{63}Ni	100	700	Yes	Yes	Water: $7x10^{-4}$; Air: $4xE^{-7}$	600	300
^{60}Co	5.27	Operational considerations apply	Yes	Yes	Water: $1x10^{-6}$; Air: $4x10^{-8}$	30	20
^{152}Eu	13.5	100	Yes	Yes	Water: $6x10^{-7}$; Air: $2x10^{-8}$	5	3
^{137}Cs	30.2	44	Yes	Yes	Water: $1x10^{-6}$; Air: $6x10^{-8}$	100	50
^{90}Sr	28.8	150	Yes	Yes	Water: $2x10^{-7}$; Air: $8x10^{-9}$	30	20
^{239}Pu	24,100	0.1	Yes	Yes	Water: $3x10^{-7}$; Air: $2x10^{-11}$	0.6	0.3
^{241}Am	432	0.1	Yes	Yes	Water: $3x10^{-7}$; Air: $2x10^{-11}$	0.5	0.2

The relevant requirements from 10 CFR Part 20 and 10 CFR Part 50 are provided in Table 4.

Table 4. Requirements of 10 CFR Part 20 and 10 CFR Part 50

Regulatory Topic	**10 CFR Part 20, Radiation Protection**	**10 CFR Part 50, Decommissioning**
Public dose limit	20.1301 (Subpart D) ≤ 0.1 rem (1 mSv) TEDE/year to the public	50.82(a)(6)(ii) (Subpart E) Site residual activity must result in ≤ 25 mrem (0.25 mSv) /year TEDE to critical group
Worker dose limit	20.1201 (Subpart C) ≤ 5 rem (50 mSv) TEDE/year 15 rem (150 mSv) to lens 50 rem (500 mSv) to skin/extremities	Not specifically addressed in Part 50 (relies on Part 20 limits during decommissioning activities)
Decommissioning criteria	Not applicable (Part 20 defines dose limits but not decommissioning endpoints)	50.82 (Subpart E) License termination requires residual activity ≤ 25 mrem (0.25 mSv)/year TEDE; As Low As Reasonably Achievable (ALARA) required
Radiation surveys / monitoring	20.1501 (Subpart F) Requires surveys of radiation levels, airborne material, contamination; must ensure adequate instrumentation and records	50.82(a)(9)(ii)(D) Final Status Surveys required to demonstrate compliance with license termination criteria
ALARA principle	20.1101(b) (Subpart B) Licensees must use procedures to keep doses ALARA	50.82(a)(3), (a)(6)(iii) Site residual radioactivity must be ALARA for license termination

NRC regulations stipulate that waste exceeding Class C limits is “not generally acceptable for near-surface disposal” (Austin, 1994), (NRC), (Kirk & Jacobi, 2016), (Supko & Schwartz, 2011), (State of Utah, Department of Environmental Quality, 2021), and should be disposed of in a geologic repository unless the NRC approves an alternative method. The NRC does not currently license a disposal facility for GTCC waste; instead, the DOE is mandated by law to develop disposal for GTCC LLW. Until a disposal path exists, licensees must safely store this waste. The NRC also oversees the design and certification of transportation packages (10 CFR Part 71) for radioactive materials, ensuring that any cask or package used to transport irradiated graphite meets rigorous containment, shielding, and testing requirements (e.g., Type B package certification if the material’s activity exceeds Type A limits).

However, if irradiated graphite can be categorized as LSA material, it may be shipped in industrial packages under DOT rules (discussed below).

A comparative analysis of the radiological hazard of GTCC waste is depicted in Figure 2.

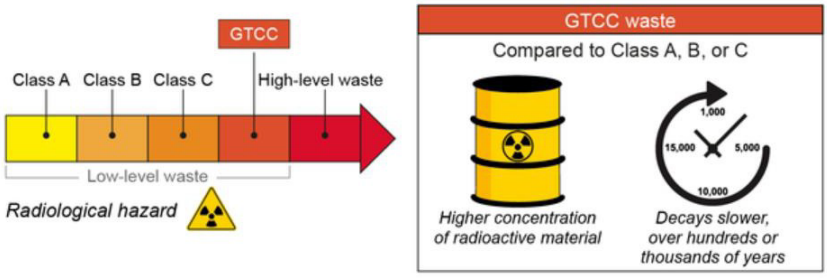

Figure 2. A Comparative Analysis of Radiological Hazard of GTCC Waste (GAO, 2022)

DOE Policies and GTCC Waste Management

The U.S. Department of Energy manages irradiated graphite waste from DOE-owned reactors and facilities, as well as the eventual disposal of commercial GTCC waste. DOE Order 435.1 establishes requirements for radioactive waste management within DOE, aligning in many respects with NRC definitions for consistency (DOE, 2021). DOE differentiates its wastes into HLW, TRU, and LLW; irradiated graphite is LLW, but if from defense programs it could be treated as TRU waste if contaminated above the TRU limits (generally not the case for pure graphite moderator material).

DOE has conducted a GTCC Environmental Impact Statement (EIS-0375) (DOE U. , 2016)(DOE, 2016), examining several disposal alternatives for GTCC LLW and GTCC-like waste, which includes irradiated reactor graphite. Alternatives evaluated include deep geologic repository disposal (e.g., in a mined repository like Yucca Mountain or Waste Isolation Pilot Plant (WIPP)), intermediate-depth borehole disposal, and enhanced near-surface disposal with robust engineered barriers. To date, no decision from the EIS has been made, and there is no operating disposal facility for GTCC waste in the U.S. As an interim measure, DOE has accepted and stored graphite waste from certain decommissioned reactors. For instance, graphite from Fort St. Vrain (a 330 MW HTGR in Colorado) and from BGRR is stored at DOE sites pending disposal. DOE's Nevada National Security Site (NNSS) has disposed of some defense-related low-level graphite waste (as seen with BGRR shipments) under DOE authority, but disposal of commercial reactor graphite would require DOE to take title. In summary, DOE policy currently favors safe storage of irradiated graphite waste at secure sites until a disposal facility is available, rather than pursuing large-scale graphite treatment, as noted by an IAEA report (IAEA, 2024). As an example, Windscale AGR irradiated graphite is now stored in steel containers in a building across the road from the original reactor.

DOT Transportation Regulations

Safe transportation is a requirement when irradiated graphite from decommissioned reactors is shipped offsite for treatment or disposal. In the U.S., the DOT (in 49 CFR 173 and 49 CFR 172) classifies radioactive material for shipment. LSA material is a category for material with limited activity concentration that is not fissile (or excepted) (NRC, 2024).

Per 49 CFR 173.403 and IAEA SSR-6 Irradiated graphite, being an "activated material" in solid form, can often be classified as LSA-II or LSA-III provided it meets the specific activity limits. LSA-II allows solids with activity up to $10^{-4}A_2$ per gram distributed throughout; LSA-III covers solid objects with essentially uniform distributed activity that are relatively insoluble (e.g., reactor components), with an average specific activity limit of $2\times10^{-3}A_2$ per gram. ^{14}C has an A_2 value (for non-special form) of five Ci (per 49 CFR 173.435), so $10^{-4}A_2$ per gram equates to 5×10^{-4} Ci/g (approximately 1.85×10^{7} Bq/g). Typical specific activity

of ^{14}C in reactor graphite can range from 10^5 to 10^6 Bq/g (depending on reactor history), which is on the order of 10^{-6} to 10^{-5} Ci/g – generally within LSA-II limits.

Thus, many irradiated graphite shipments have been made as LSA material in strong, tight containers. For example, during the BGRR decommissioning, graphite blocks were placed into robust steel containers (B-25 boxes or similar[4]) and then into intermodal freight containers for rail/truck shipment. These would have been certified to meet IP-2 standards required for LSA-II material. DOT rules also impose limits on radiation levels (e.g., <10 mSv/h at package surface for exclusive-use LSA shipments) and external contamination. In cases where graphite exhibits higher-activity concentration or fails LSA criteria (e.g., small high-activity components), a Type B cask approved by the NRC would be required for transport, adding complexity. Overall, DOT and parallel IAEA transport regulations ensure that irradiated graphite can be transported safely under either the LSA framework for bulk shipments or with Type B certified packaging if needed. These regulations are an integral part of the waste management process, bridging the gap between decommissioning sites and disposal or treatment facilities. In some other countries (e.g. Germany) irradiated graphite is additionally enveloped in plastic bags to avoid dust releases.

IAEA Graphite Waste Classification

The IAEA classifies radioactive waste based on disposal requirements rather than concentration limits. This system, outlined in (IAEA, 2024), (IAEA, 2009) and (IAEA, 2011), provides greater flexibility for managing irradiated graphite. Key IAEA categories, listed in Table 5, include:

Table 5. IAEA Waste Classification and Description

Waste Classification	Description and Disposal Path
Very Low-Level Waste (VLLW)	Minimal radiological hazard, suitable for controlled near-surface disposal.
LLW	Short-lived radionuclides, requiring engineered barriers in near-surface disposal.
ILW	ILW contains higher amounts of radioactivity than LLW but does not generate significant heat (i.e., thermal output is low and does not require heat management in disposal). Requires engineered disposal, typically in shallow to deep geological facilities, but not necessarily deep geological isolation.
High-Level Waste (HLW)	Requires deep geological disposal due to heat generation; rarely applies to graphite.

IAEA does not mandate regulations but provides guidance, safety standards, and a forum for international best practices. In the context of irradiated graphite, IAEA TECDOC publications (e.g., (IAEA, 2011),

[4] A B-25 box is a standardized steel container widely used for the packaging, storage, and transportation of low-level waste and other hazardous materials. It is commonly employed in nuclear decommissioning projects, including graphite waste disposal, because of its durability and regulatory compliance. Key Features of a B-25 Box: Material: Carbon steel (typically 12-gauge, ~2.7 mm thick), Dimensions: 4 ft × 4 ft × 6 ft (approx. 1.2 m × 1.2 m × 1.8 m), Volume Capacity: ~90 cubic feet (~2.55 cubic meters), Weight Capacity: ~4,000 - 6,000 lbs (~1,800 - 2,700 kg), depending on design. Lid Type: Can be bolted, welded, or equipped with a gasketed lid for added containment. Stackability: Designed to be stackable for efficient storage and transport.

(IAEA, 2016)), and coordinated research projects have been pivotal. The IAEA's International Project on Irradiated Graphite Processing Approaches (GRAPA) compiled strategies from numerous member states on graphite waste management (Wickham, Meyer, & O'Sullivan, 2019). A key observation is that approaches differ widely: some countries (e.g., France and Lithuania) opt for immediate dismantling of graphite reactors and prompt waste packaging, while others (e.g., U.K.[5], Russia, and USA) have until now favored deferred dismantling and long-term safe storage of graphite in situ, awaiting future disposal solutions. Waste acceptance criteria (WAC) for disposal of graphite also vary. For example, Spain has developed criteria to dispose of reactor graphite as LLW in its El Cabril near-surface repository, by considering pre-treatments to ensure that the graphite meets the prescribed limits for long-lived radioisotopes. Conversely, France classifies reactor graphite as waste requiring deep geological disposal (as part of the ILW inventory[6] for its planned repository) due to its high long-lived activity (Pina, 2021), (OECD/NEA, 2014), (Fanny, 2013), (Andra, 2015). The IAEA promotes thorough characterization of graphite (radiological and physical) to underpin whatever management route is chosen. It notes that "the characterization of irradiated graphite (modeling as well as analytical) to determine the distribution and quantity of radioactivity is well established" (IAEA, 2024) and is essential for selecting dismantling or treatment options.

International transport of radioactive graphite falls under the IAEA's Regulations for the Safe Transport of Radioactive Material (SSR-6) (IAEA, 2018), which are reflected in DOT and NRC rules. For instance, graphite blocks or pieces with distributed activity[7] may qualify as LSA-II or LSA-III material (depending on the uniformity and specific activity[8]) per IAEA/DOT definitions, allowing shipment in IP-2 or IP-3 industrial packages with appropriate labeling and exclusive-use provisions, rather than requiring a hardened Type B cask for each shipment. More discussion on this classification is made in Section 4.6. IAEA guidance emphasizes safety, environmental protection, and the use of engineered and administrative controls for storage and disposal of graphite waste, encouraging information exchange on emerging technologies.

Salient aspects of the synergies and differences between the NRC regulatory classification and the IAEA LLW classification are provided in Table 6.

---

[5] The information in Table 5. IAEA Waste Classification and Description, however, does not address concerns about loss of knowledge, together with a desire to re-use nuclear-licensed and grid-connected sites. Thus, currently Nuclear Restoration Service (NRS) is actively supporting research to address the above concern for both Magnox and AGRs. The NDA baseline policy for graphite is to place the waste in a geologic repository when one exists (which implies a long deferral), but they are open to new initiatives through the NRS program

[6] U.S. Nuclear Regulatory Commission (NRC) does not have an "intermediate-level waste" classification. However, in the NRC's system, waste that might be considered ILW internationally often falls into Class C LLW or GTCC waste.

[7] Refers to the spatial distribution of radioactive contamination within the graphite structure. It describes how radioactivity is spread throughout the bulk of the graphite, whether it is uniformly distributed or concentrated in specific areas. The distribution can vary due to factors like neutron flux gradients, impurity levels, and graphite porosity. In nuclear reactors, neutron activated radioisotopes like ^{14}C and ^{3}H are embedded within the graphite structure.

[8] Defined as the radioactivity per unit mass of the material, typically expressed in Becquerels per gram (Bq/g) or Curie per gram (Ci/g). It quantifies the intensity of radiation emitted by a given mass of graphite waste.

Table 6. A Comparison of the NRC and IAEA LLW Waste Classifications

Criteria	NRC (U.S.)	IAEA (International)
Main regulatory framework	10 CFR Part 61 & 72	IAEA GSG-1 & SSR-5
Primary graphite classification	GTCC Waste (most cases), some LLW (Class C)	ILW
^{14}C handling	Strict disposal limits; often exceeds Class C	Managed within ILW; flexible engineered disposal
Disposal pathway	GTCC graphite requires deep geological disposal	ILW graphite can use deep boreholes or near-surface engineered storage
Flexibility in classification	Strict, concentration-based limits	Disposal-based approach allows flexibility

The differences in the classifications shown in Table 6 have the following implications for harmonization. The current stricter NRC classification (GTCC) results in more limited disposal pathways compared to IAEA's ILW category. IAEA's approach allows some near-surface disposal, while the U.S. requires deep geologic disposal for GTCC graphite (NEA, 2016).

A brief discussion of the challenges involved in international harmonization of rules and regulations with respect to irradiated graphite waste management is provided in Appendix D.

U.K. Waste Classification and Management

In the U.K., an overarching waste management principle informs the development and implementation of the waste management (NDA, 2020). The use of the waste hierarchy is intended to drive waste prevention and minimize overall volumes where possible.

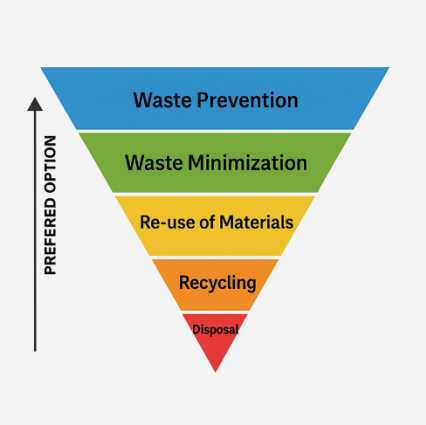

Figure 3. British Practice of Waste Management Hierarchy

In the U.K., radioactive waste is classified according to the nature and quantity of radioactivity and the heat produced. The categories stated in Table 7 are used.

Table 7. U.K.'s Waste Classification

Waste Category	Description	Significance
HLW	Temperature may rise significantly because of its radioactivity.	Thermal effects should be accounted for in the design of storage or disposal facilities.
ILW	Waste exceeding the upper boundaries for LLW. Does not generate sufficient heat.	Thermal considerations may not be an issue to be considered in the design of storage or disposal facilities.
LLW	Waste with a radioactive content not exceeding four Gigabecquerels per tonne of alpha activity, or 12 Gigabecquerels per tonne of beta/gamma activity.	
VLLW	A sub-category of LLW.	It can be safely disposed of alongside municipal, commercial, or industrial waste, or can be disposed of in specified landfill sites, subject to limits on radioactivity content.
Higher-Activity Waste (HAW)†	Encompasses HLW, ILW and a small fraction of LLW.	
Lower Activity Waste (LAW)	Encompasses both LLW and VLLW.	

† The term HAW refers to all radioactive material that has no further use that falls into the following categories: HLW, ILW and the relatively small volume of LLW that is not deemed suitable for disposal at the LLWR or the LLW facility at Dounreay. NDA strategy involves converting the HAW within the NDA estate into a form that can be safely stored and managed.

The strategic positions and preferences of the NDA, as of 2020, are summarized in Table 8.

Table 8. Summary of Strategic Positions/Preferences

Regulatory Guidance	Waste Category	Disposal Capability	Disposal Capability Owner
Radioactive wastes – guidance requirements for authorization	ILW (noting that not all ILW will be suitable for disposal by this option)	Scottish HAW disposal option (planned future capability)	To be determined
	LLW/ILW	Near-surface disposal (England/Wales) (potential future capability)	

Regulations and Design Criteria for Irradiated Graphite Management in Other Countries

The irradiated graphite waste management from HTGRs and MSRs has been addressed by Japan, China, and South Korea by using regulations, and design criteria for waste containers.

The regulatory framework for Japan is governed by Nuclear Reactor Regulation Law under the Nuclear Regulation Authority (NRA). Irradiated graphite is classified as an ILW. The storage and transport requirements follow IAEA SSR-6 with additional national seismic and moisture control standards.

China classifies irradiated graphite according to Chinese GB 11806 Standard as ILW[9]. The storage and transport requirements comply with this Standard and are aligned with IAEA SSR-6, with special graphite waste provisions. Irradiated waste management in South Korea is governed by the Nuclear Safety Act, enforced by the Nuclear Safety and Security Commission (NSSC). Per KINS Notice 2014-02[10], irradiated graphite waste is classified as ILW and is managed. The storage and transport requirements are aligned with IAEA SSR-6, with emphasis on graphite-specific dust management.

Irradiated graphite waste in France is regulated by Agence Nationale pour la Gestion des Déchets Radioactifs (ANDRA) and Autorité de Sûreté Nucléaire (ASN). Irradiated graphite is classified as VLLW or LLW, requiring near-surface disposal. The Cigéo project aims to establish deep geological repositories for long-lived radioactive waste. In Germany, radioactive waste management is governed by the Federal Office for the Safety of Nuclear Waste Management (BASE) and Bundesamt für Strahlenschutz (BfS). Irradiated graphite is classified as negligible-heat generating ILW and HLW depending on contamination level, such as the baked-carbon structures of AVR, exceeding the WAC of the KONRAD repository. Storage and transport follow IAEA SSR-6 with strict adherence to deep geological disposal planning, emphasizing high-integrity shielding.

In the U.K., the oversight for the management of irradiated graphite waste is by the Office for Nuclear

[9] Specific details on how irradiated graphite is classified under GB 11806 are not readily available.

[10] Not available at the Korea Safety Authority website but mentioned in a CARBOWASTE document mentioned previously.

Regulation (ONR) while still on the licensed reactor site, and thereafter of the Environment Agency (England and Wales) or the Scottish Environmental Protection Agency. Irradiated graphite sleeves from AGRs are already stored at Sellafield, while some Magnox-reactor fuel-element components remain in storage vaults onsite. The long-term default plan is for deep disposal: however, strategies are under review with a possible future focus on conditioning, treatments, and possible recycling of valuable isotopes and reuse in new graphite manufacturing for the nuclear industry.

In Spain, management oversight for irradiated graphite waste is the responsibility of Empresa Nacional de Residuos Radiactivos (ENRESA). Spain utilizes El Cabril near-surface repository for LLW disposal. However, research into alternative treatment methods continues. In Lithuania, the irradiated graphite waste management is regulated by the State Nuclear Power Safety Inspectorate (VATESI). The decommissioning of Ignalina Nuclear Power Plant (NPP) involves a major irradiated graphite disposal challenge. Lithuania is considering deep geological disposal for long-term safety. A summary of the status is provided in Table 9.

Table 9. Irradiated Waste Classification, Storage, and Disposal Strategy for Selected Nations

Country	Classification	Storage Method	Disposal Strategy
Japan	LLW	Interim storage, monitored	Geological disposal
China	LLW/ILW	Temporary storage	Deep disposal under review
South Korea	LLW	Engineered repository	Near-surface disposal
Germany	ILW (negligible heat generation)	Interim storage	Geological disposal
Germany	HLW	High-integrity concrete shielding, multilayer containment, special gas sealing	Strict long-term disposal strategy, focus on deep geological storage and retrievability
France	VLLW/LLW	Surface storage	Near-surface & geological disposal
U.K.	LLW/ILW	Site-specific storage	Future deep disposal (baseline strategy) if alternative solutions remain unviable. For Scotland, shallow burial adjacent to source sites is the baseline strategy.
Spain	LLW	Near-surface repository	El Cabril site Not all irradiated graphite qualifies as LLW under Spain's criteria. It must undergo characterization to confirm it meets LLW limits for long-lived radionuclides like ^{14}C, ^{36}Cl, or ^{3}H before acceptance. (Grambow, et al., 2013)
Lithuania	ILW	Temporary storage	Geological disposal under study, See Table B-1 and Table L-6 in (VATESI, 2024)

The design criteria for waste containers are briefly summarized in Table 10.

Table 10. Design Criteria for Waste Containers for Various Select Countries

Country	Container Features	Special Considerations
Japan	Double containment, 800 °C fire resistance, seismic qualification	Tritium release and moisture control
China	Multilayer shielding, nitrogen atmosphere, dust control systems	Graphite dust mobility and early characterization
South Korea	Double containment, real-time dust monitoring, shock absorbers	Salt compatibility, chemical decontamination (for MSRs)
Germany	High-integrity concrete shielding, multilayer containment, special gas sealing	Strict long-term disposal strategy, focus on deep geological disposal

The Japanese experience consists of HTTR operations and waste stream characterization by the Japan Atomic Energy Agency (JAEA). It was found that preventing significant tritium release from irradiated graphite requires improved moisture and ventilation controls (Kawamura et al, 2021). Tritium formation and mitigation have also been studied extensively (Schmutz, Sabharwall, & Stoots, 2012), including reactions in MSRs. From German experience (Wahlen, Wahl, & Pohl, 2000), it can be concluded that fine graphite dust can be expected to present sealing challenges in waste containers. The Chinese experience in graphite handling and disposal strategy development is related to HTR-10 and HTR-PM pebble-bed reactors. The lessons learned were: (a) early underestimation of graphite dust mobility led to revised transport protocols (Zhang et al., 2014), and (b) an emphasis on early waste characterization improved container design (Li et al., 2015).

Korea Atomic Energy Research Institute (KAERI) research on Very High-Temperature Reactor (VHTR) and MSR graphite waste handling provide experience in South Korea. The lessons learned include: (a) real-time dust monitoring significantly reduced contamination risks (Park et al., 2017), and (b) the MSR-irradiated graphite required specialized pre-conditioning for salt removal (Kim et al., 2018).

Germany has managed irradiated graphite from decommissioned research reactors. After the AVR reactor was shut down in 1988, the dismantling of its highly radioactive graphite core internals posed significant technical challenges due to high levels of activation (notably ^{14}C and tritium) and fission product contamination. No graphite blocks or baked-carbon core components were removed from the AVR reactor vessel. Instead, the entire reactor pressure vessel (RPV), containing the graphite internals and residual graphite dust, was filled with lightweight concrete to immobilize its contents. The sealed RPV was then transferred as a single unit to long-term interim storage at the nearby Jülich site (Sterner & Rittscher, 2009). Due to ongoing public and regulatory scrutiny over long-term safety, the storage strategy remains a topic of debate, particularly in relation to ultimate disposal options and transport into the Ahaus interim storage. The transfer of about 630000 irradiated fuel, moderator and absorber pebbles from THTR into the Ahaus interim storage was managed within schedule (Schneider-Eickhoff R. and Stuke M., 2024). The shipment was performed by train in 305 CASTOR casks with usually six casks on two railway wagons. The maximum processing rate was 11 CASTOR casks per week in 3-shift operation, 6 days a week. (Plätzer S. and Mielisch M., 1998).

Figure 4: Shipment of THTR Spent Fuel, Graphite and Absorber Pebbles in CASTOR Canisters

To summarize the multinational lessons learned, it behooves to pay attention to: (a) applying multilayer containment with real-time dust and gas monitoring; (b) developing pre-conditioning steps tailored to reactor type (e.g., salt decontamination for MSRs, surface decontamination); (c) performing comprehensive early characterization of irradiated graphite to inform container design; (d) enhancing regulatory integration of graphite-specific properties (dustiness, tritium/radiocarbon release); (e) determination of organic and inorganic nature of radioactive releases, and (f) adopting high-integrity engineered and geological shielding approaches (e.g., used in Finland and Germany) for long-term disposal.

A Summary of the International Management of Irradiated Graphite Waste

The management of irradiated graphite waste is a complex challenge faced by many nuclear programs around the world. This section summarizes the approaches adopted in the United States, Spain, Lithuania, China, Korea, and Japan. It highlights the prevailing international trend to avoid incineration for irradiated graphite, primarily due to the technical, regulatory, and economic challenges associated with ^{14}C release.

In the United States, irradiated graphite has consistently been managed through storage, conditioning, and direct disposal rather than incineration. The DOE has historically favored encapsulation and geological disposal (Mueller, Steele, & Gellender, 2019). Studies by the (IAEA, 2004) confirms that U.S. policy emphasizes non-thermal methods, with incineration being considered neither practical nor cost-effective for large graphite inventories. (IAEA, 2004) discusses U.K. industry evaluations of potential encapsulation matrices for irradiated graphite. It specifically notes (on page 43) that: “It is of note that no mention is made of the potential application of incineration or other volume-reduction techniques as a graphite waste-management tool.” It confirms that the omission of incineration or thermal volume reduction methods was intentional, deeming them impractical or undesirable for managing large graphite inventories at that time.

Spain’s approach to irradiated graphite from the Vandellòs I reactor highlights similar conclusions. Fuel-

element graphite from this decommissioned reactor is packaged and stored for eventual disposal: the moderator and reflector remain in the pressure vessel. Incineration was not pursued due to concerns over ^{14}C release and lack of regulatory acceptance (IAEA, 2006), (Enresa, 2020) but thermal treatment with the objective of reducing the specific activity of weak beta-emitting isotopes is under serious consideration.

Lithuania's Ignalina NPP is another example where immediate removal of graphite from the reactor cores has been prioritized. Decommissioning plans focus on temporary storage and later disposal in a deep repository, with no active consideration of incineration due to cost, technical issues, and regulatory challenges (IAEA, 2025) (IAEA, 2004)[11]. China has explored thermal treatment technologies at a research level; however, these efforts remain limited and have not yet led to operational deployment. Current policy favors conditioning and disposal, consistent with the international trend (Deng, Wu, Xie, & Li, 2008). In Korea, irradiated graphite primarily comes from research reactors. KAERI evaluated thermal options but determined that direct disposal is preferable based on cost, regulatory simplicity, and technical feasibility (KAERI, 2015). IAEA TECDOC-2072 (IAEA 2024, Annex IV) documents that most national graphite waste programs—including Japan's—do not use thermal treatment, opting instead for onsite conditioning (e.g., cementation, immobilization) followed by disposal in repository facilities. Germany has placed the whole AVR RPV housing the complete graphite and baked-carbon components into long-term interim storage. The decommissioning of the THTR being under 'safe enclosure' is currently being prepared but dependent on the operation of the KONRAD repository.

[11] Currently, two independent consortia have developed detailed plans for removal of the entire high-active cores (so-called R3 zone'), and their options are under consideration by the Lithuania Ministry of Energy and the European Commission (the funding body).

5. IRRADIATED CARBONACEOUS MATERIAL BEHAVIOR AND TREATMENT

Radionuclide Inventory and Distribution: The radiological characteristics of irradiated graphite, graphitic matrix and baked carbon determine both their hazard profile and the strategies for their management, including any treatment. Irradiated graphite is more intensively investigated and submitted to treatment than other forms of irradiated carbonaceous waste. Thus, the focus is put on irradiated graphite within this and the following chapters.

As noted, the dominant long-lived radionuclides are ^{14}C and ^{36}Cl, produced by activation of nitrogen or carbon and chlorine impurities, respectively. ^{14}C is usually the single largest contributor to long-term radioactivity in graphite, often accounting for > 90% of the total activity after a few decades of decay. ^{36}Cl (301,000-year half-life) is present in smaller absolute amounts but is highly mobile in groundwater, thus important in safety assessments. Tritium (half-life 12.3 years) is a significant contributor to the initial radioactivity: although tritium decays to negligible levels over a few centuries, tritium radioactivity is a serious consideration during decommissioning and storage due to its high specific activity and volatility. ^{60}Co (half-life 5.3 years), resulting from trace cobalt in graphite or alloy fittings, is often the primary gamma emitter initially, elevating dose rates from freshly discharged graphite. However, ^{60}Co decays away to safe levels within a few decades of interim storage. Other important isotopes include ^{90}Sr and ^{137}Cs, if fuel failures caused contamination, and trace alpha-emitters (e.g. uranium, plutonium, and americium) from uranium fuel contamination within the graphite components and carbon matrix of spherical fuel elements or fuel compacts or uptake of coolant-born species. Most of the above isotopes can be significantly reduced or even removed from the waste stream (e.g. tritium) by thermal and chemical treatment processes. ^{14}C is a special challenge, as it is chemically reacting identically with natural ^{12}C and ^{13}C. Typically, the total transuranic content in reactor graphite is extremely low (less than 0.01% of activity), meaning the waste does not qualify as TRU waste.

The distribution of these radionuclides in graphite is not uniform: fission products or contamination may be almost exclusively in the surface layers, while ^{14}C may be generated both at the surface and within the bulk material, via ^{13}C activation. The ^{14}C may, however, also be formed preferentially at the surface, as will be explained. Measurements have shown a remarkable aggregation of ^{14}N at the surface of graphite samples even after conditioning under vacuum for three days (Vulpius D. et al. 2013): The radionuclides investigated in this study can be divided into three parts: tritium, radiocarbon and metallic activation and fission products. Tritium can be bound in neutron-irradiated nuclear graphite as strongly adsorbed tritiated water (HTO), in oxygen-containing functional groups (e.g. C–OT) and as hydrocarbons (C–T). Radiocarbon is covalently bound with the graphite structure. The activity can be described by a homogeneously distributed part and a heterogeneously distributed part (enriched on surfaces or in hotspots). Metallic radionuclides can be bound as ions or covalent metal–carbon compounds. The distribution of all these radionuclides is mainly dependent on the distribution of their inactive precursors. These findings were evidenced at irradiated AVR and St. Laurent graphites. These scientific interpretations are supported by a pronounced ^{14}C surface contamination, which has been observed on spent fuel pebbles of AVR and the fuel spines of Peach Bottom Unit 1. Advanced analytical techniques (e.g., drilling sampling, leaching studies, autoradiography) and modeling have been used to map radionuclide distribution, which is crucial for designing decontamination or encapsulation methods. Clusters of activated impurities have been observed by radiographies within the irradiated graphite of different origins (Kuhne, Lokalisation, Freisetzung, & Speziesbestimmung, 2017). Their study also observed that hydrophobic virgin graphite shows hydrophilic behavior after neutron irradiation. This should especially be considered when assessing storage and disposal features of irradiated graphite.

Purified irradiated graphite may also be reused as a resource for manufacturing new components, such as monolithic molded reflector or fuel blocks, for example by applying the technology of A3 matrices (M. Hrovat et al. 2008).

For reuse or other forms of waste management, various volume reduction and treatment methods could be utilized, including oxidation, thermal treatment, other chemical treatments (including electrochemical treatments), and/or microbial exposure. A treatment facility that handles irradiated graphite and/or fuel elements would require licensing according to nuclear regulations for both normal operation and accident conditions.

6. INTERIM STORAGE AND CONDITIONING

In the absence of immediate disposal, irradiated graphite is often placed in interim storage after retrieval and possible treatment. The waste may be conditioned by encapsulation – e.g., immobilizing graphite pieces in a cement or polymer matrix inside drums or containers – to meet stability requirements (prevent dispersal, ensure physical integrity). The NRC and IAEA require that waste intended for near-surface disposal be in a stable form to inhibit dispersal and intrusion. For graphite destined for geological disposal, such stringent conditioning may not be required, but encapsulation can still aid in handling and stacking. Interim storage of graphite waste has taken place in structures like engineered concrete vaults, metal storage casks, or existing reactor containments. In the U.K., for instance, graphite from the defueled Windscale Piles and Magnox reactors remains stored within the reactor pressure vessels. During dismantling of the BGRR, graphite blocks were first placed into soft-sided containers ("super sacks") and then loaded into metal shipping containers—this packaging was done onsite, and the blocks remained there for a period before being shipped off-site (Kirby, 2011). Fort St. Vrain's graphite was loaded into canisters and stored at Idaho National Lab's waste facilities. The technical considerations for storage include ensuring the graphite is dry (to avoid any corrosion or radiolytic generation of organic ^{14}C bearing gases), vented if necessary to prevent pressure buildup (graphite outgassing is low but nonzero), and monitored for any dose rate changes. Graphite's stability (it does not corrode like metals or degrade like organic resins) makes it a generally stable waste form for storage, which is an argument some have made for leaving it unprocessed. Note however that there are some concerns about the establishment of electrochemical cells when graphite is stored in steel containers, potentially leading to contained corrosion if moisture is present. There has been opinion that graphite itself is an excellent waste form for containing embedded radionuclides, so removing them might only be justified if it significantly eases disposal classification or enables recovery of a useful isotope such as ^{14}C.

Microbial attack is another issue to be addressed because experiments with exposure to specific bacteria have shown a relevant release of ^{14}C (Dunzik-Gougar 2008).

The experimental results of the interim storage of spent AVR fuel are presented in Appendix F.

Long-Term Behavior Modeling

Advanced modeling plays a role in understanding graphite waste in a repository environment. Performance assessment models simulate how ^{14}C and other isotopes might be released over thousands of years. Graphite's porosity and the chemical form of ^{14}C influence release: ^{14}C can exist as organic functional groups on the surface and in pores or as part of the lattice. Leaching experiments under simulated repository conditions show extremely low release rates. In one study on TRIGA[12] reactor graphite, after one year of leaching under anaerobic conditions, only ~1.85% of the initial ^{14}C inventory leached out into solution (Bucur, Ichim, & Florea, 2018). Under aerobic conditions, a similarly low percentage (~ 1.75%) was

[12] Training, Research, Isotopes, General Atomics (TRIGA). TRIGA reactor was developed by General Atomics in the late 1950s. The Mark I model began operation in 1958 in San Diego, and over 66 units have been built worldwide in 24 countries. Uses an innovative uranium-zirconium hydride (UZrH) fuel that has a prompt negative temperature coefficient: as the fuel heats up, the reactivity sharply drops, causing automatic shutdown. This unique property allows safe pulsing from near zero up to tens of GW for milliseconds—or even up to ~22 GW in some tests—without fuel damage (General Atomics, n.d.), TRIGA International, a joint venture company with CERCA of France, manufactures and sells TRIGA fuel to research reactors.

released, though the speciation of ^{14}C differed (aerobic leaching produced more inorganic carbon, while anaerobic produced more organic compounds like methane). These results reinforce that graphite is a very robust matrix: the vast majority of ^{14}C remains bound even when submerged in water for long periods, which is encouraging for long-term disposal safety. However, potential releases >1% (volatility) could already alter the waste classification e.g. under German WAC for the KONRAD waste repository (Brennecke et al. 1987).

Models incorporate such leach rates and the diffusion of gaseous ^{14}C (as $^{14}CO_2$ or $^{14}CH_4$) in evaluating potential doses. Another aspect is the effect of graphite radiolysis: in radiation fields, graphite can produce small amounts of methane, hydrogen, or carbon oxides. In a sealed repository, due to its energetic nature buildup of $^{14}CH_4$ could pose a containment concern if not accounted for, but studies (e.g., in France's disposal research) indicate the amounts are limited by graphite's low radiolytic gas yield.

In this survey of graphite waste regulations, a brief analysis was conducted on the decay characteristics of radionuclides that influence the temperature and pressure over long-term storage. The results are presented in Appendix D.

Cask and Package Design Considerations test

When irradiated graphite waste is packaged for transport or disposal, licensees must ensure the packages meet structural, thermal, shielding, dust retention and containment requirements. Graphite waste emits primarily beta radiation (from ^{14}C, ^{3}H) and some gamma (from ^{60}Co, ^{152}Eu, etc., if present). For shielding design, a few centimeters of steel or concrete can easily attenuate gamma radiation to acceptable levels from typical graphite waste, and beta radiation is stopped by the concrete[13], which is a low Z material. The more challenging issues are often contamination control and containment of any volatile radionuclides. Cask designers examine whether a graphite waste cask might accumulate any pressure from gas release. If graphite is stored in an air atmosphere radiolysis could generate hydrogen and ^{14}CO, $^{14}CO_2$, and possibly organic compounds within the sealed canister. Consequently, many waste packages for graphite include filtering vents or are not sealed to hermetic levels required for HLW. For example, an industrial package might have a filtered vent to prevent pressurization while still trapping particulate. Another challenge considered was the potential presence of Wigner energy in graphite from certain reactors, e.g., the 1957 Windscale pile fire was caused by uncontrolled Wigner energy release. Any graphite retrieved from a low-temperature reactor that has not been annealed could, in theory, release heat suddenly if heated or mechanically shocked. Operators planned (and equipped facilities) to anneal most Pile 1 graphite before packaging, but there is evidence the blocks were ultimately packaged without annealing. However, in many cases the degree of stored energy will be saturated (formation rate equaling the combined rates of thermal and radiation annealing) - this is the case with Magnox reactors, for example. Going forward, the vast majority of graphite from HTGR or MSR will not have Wigner energy and thus the possibility of an uncontrolled Wigner energy release should be deemed to be extremely unlikely.

As graphite is a neutron moderator, even though irradiated graphite waste itself is not fissile, if large amounts are packaged near fissile material, it could potentially affect criticality safety. In transportation, the criticality safety index (CSI) of packages assumes a certain configuration – with graphite waste, this is usually not relevant unless mixed with sufficient fissile material, which is generally not allowed for LSA material

[13] Concrete is a heterogeneous mix of low-Z elements—hydrogen, oxygen, silicon, calcium, aluminum, etc.—plus small amounts of iron and sodium. Its overall effective atomic number (Z) is ≈ 7–11.

beyond exempt limits. However, for spent fuel casks that include graphite, such as sleeves or blocks co-disposed of with fuel (e.g. direct disposal of spent HTGR fuel), NRC staff may need to verify that the graphite does not undermine the package's criticality control design. There is ongoing technical discussion about co-disposal of graphite and spent TRISO fuel. One option is to package graphite fuel blocks together with fuel compacts in dual-purpose canisters. Analyses have shown that it is feasible but could increase package volume and require careful gas generation analysis (due to graphite radiolysis and the presence of zirconium or other reactive components, e.g., uranium carbide (UC) kernels in TRISO fuel.

In summary, the technical analysis indicates that irradiated graphite waste can be safely handled, transported, and stored with existing technology. The graphite strongly retains radionuclides (especially ^{14}C) under normal conditions, which is favorable for long-term storage or disposal. A variety of treatment options exist to reduce volume or remove problematic isotopes, each with advantages and trade-offs. The regulatory framework provides clear classification and design targets, but the lack of available disposal facilities for long-lived graphite waste (GTCC/ILW) and insufficient experimental validation of graphite (and matrix) samples irradiated under HTGR and MSR conditions means interim measures and novel solutions are needed. The results of modeling and experiments so far support the safety of extended storage and inform the design of future deep repositories to accommodate graphite. The following discussion (Section 7) will integrate these findings and address the practical and regulatory challenges that remain.

7. CHALLENGES IN HARMONIZING TRANSPORTATION AND STORAGE OF GRAPHITE WASTE FROM HTGR AND MSR

This section discusses non-technical challenges associated with harmonizing the transportation and storage of irradiated graphite (i-graphite) waste from HTGR and MSR. Both reactor types are under development in the United States as part of the next generation of advanced nuclear technologies. Effective management of i-graphite waste is crucial to ensure regulatory compliance, environmental safety, and public confidence.

Regulatory Harmonization Across Jurisdictions: HTGR and MSR designs are under development not only in the U.S. but also internationally (e.g., China, U.K.). This creates challenges in aligning U.S. NRC transport and storage regulations with international standards. Harmonized approaches could reduce regulatory burdens on developers and facilitate cross- border technology transfer (NEA, 2012).

Public Perception and Stakeholder Engagement: Public perception of irradiated graphite, particularly due to concerns over ^{14}C, requires proactive risk communication. Establishing clear, science-based messaging and opportunities for public engagement will be essential in obtaining transport and storage approvals.

Economic and Policy Uncertainty: Since HTGR and MSR deployment timelines remain uncertain, investment in dedicated transport and storage infrastructure for i-graphite involves financial risk. Economic risk will be reduced if the NRC, DOE, and industry coordinate to ensure both industry and regulators are prepared for future waste management strategies (NEA, 2022).

International Collaboration and Data-Sharing: Most existing data on irradiated graphite comes from legacy gas-cooled reactors such as Magnox and AGR. New HTGR and MSR designs introduce graphite variants with distinct irradiation and chemical histories. International collaboration and data-sharing agreements will be of great benefit to develop robust U.S. regulatory criteria for graphite waste characterization and packaging (NEA, 2016).

A summary of challenges that should be addressed is provided in Table 11.

Table 11. Summary of Challenges to Address

Challenge Type	Key Issues	References
Characterization	Radionuclide inventory, physical variability	NEA (2016), IAEA (2006)
Stability	Dust generation, gas buildup	NRC (2013), NEA (2016)
Packaging	Shielding, volume optimization	IAEA (2018), NEA (2016)
Disposal compatibility	Waste acceptance criteria	NEA (2016), NRC (2019)
Regulatory	Cross-border alignment	NEA (2012)
Public perception	Risk communication	NRC (2019)
Economic uncertainty	Infrastructure planning	NEA (2022)
Data-sharing	Access to operational data	NEA (2016)

Case Studies and International Comparisons

U.K. - Management of Graphite Waste from Magnox and AGR Reactors: The United Kingdom has substantial experience managing irradiated graphite from its fleet of Magnox and AGR. The U.K.'s strategy

combines onsite interim storage with research into disposal pathways for long-lived isotopes like ^{14}C. Key lessons include the importance of early waste characterization, public transparency, and developing bespoke conditioning processes to reduce dust formation and radiolytic gas generation. The U.K.'s approach is overseen by the Environment Agency and the Nuclear Decommissioning Authority: on the latter's behalf, the recently formed Nuclear Restoration Services is charged with accelerating dismantling to enable reuse of sites, and innovative removal and treatment of the irradiated graphite is very much on its agenda. This experience provides useful reference points for NRC regulatory strategy development (NEA, 2016).

France: In France, graphite waste comes primarily from the early generation of gas-cooled reactors (UNGG reactors). The French waste management agency, ANDRA, has explored both near-surface and deep disposal options, but faces ongoing challenges related to the long-lived ^{14}C inventory and its potential migration in repository environments. This highlights the importance of coupling transport and storage regulations with clear, science-based disposal criteria (NEA 2012).

International Collaboration: International forums like NEA's Irradiated Graphite Working Group and IAEA Technical Meetings provide vital platforms for harmonizing approaches to graphite waste management. For the NRC, leveraging these platforms can support the development of guidance that is both internationally aligned and tailored to the unique conditions of U.S. advanced reactors. Lessons from countries with extensive graphite decommissioning experience—such as the U.K., France, and Germany—may be valuable as a means to inform U.S. regulatory development (NEA 2016).

Some aspects that would reduce regulatory risks to industry are noted below.

- Stakeholder benefit could result from early engagement between advanced reactor developers, the NRC, and DOE to proactively address waste classification and packaging.
- Stakeholder benefit could result from an expansion of publicly available data on irradiated graphite properties from international decommissioning projects to support characterization benchmarks.

Stakeholder benefit could result from international harmonization through collaboration with NEA and IAEA to avoid regulatory fragmentation for cross-border technology transfers**.**

Integration of Regulatory Requirements and Industry Practices

Managing irradiated graphite requires aligning practical treatment and storage methods with a complex regulatory environment. Compliance with NRC, IAEA, DOE, and DOT requirements is the first layer for ensuring safety. Industry's best practices for irradiated graphite waste management have evolved within these regulatory guardrails, often through operational experience in reactor decommissioning projects worldwide. A key observation is that no single management solution fits all scenarios. Differences in national policy, the radiological characteristics of the graphite, and available infrastructure lead to varied approaches.

In the United States, where irradiated graphite from commercial reactors is effectively GTCC waste, the absence of a near-term disposal site means licensees must containerize and store graphite safely onsite or at interim facilities (often DOE sites) for an extended period. This reality has driven best practices focusing on robust interim storage – for example, designing secure onsite storage vaults that can last 50+ years, in conjunction with monitoring programs for any contamination or degradation. Other countries, such as Spain, have classified some graphite as acceptable for near-surface disposal by meeting strict WAC limits (Pina, 2021).

In Spain, the best practice was thorough characterization and selective segmentation of the graphite to ensure each package meets the Class B/C equivalent limits for ^{14}C and other radionuclides. France has committed to

geological disposal for graphite along with other ILW; hence EDF's practice is to package graphite in durable containers and place them in surface storage facilities until the CIGEO repository becomes available. Deferred dismantling reduces near-term risk and allows isotope decay but transfers the burden to future generations – a trade-off that regulators and stakeholders weigh differently by country.

Transportation logistics of irradiated graphite has been honed through projects like BGRR and Fort St. Vrain (U.S.), as well as the transfer of German THTR reactor pebble waste to interim storage. A successful practice used to minimize radiation dose and risk during transport was the "Exclusive use" approach for LSA material shipments - dedicating an entire conveyance (train or truck) to the waste, which allows higher package surface dose rates (still within regulatory limits) and minimizes handling en route. For international shipments, Type B containers may be needed (for example, if shipping spent fuel pebbles from Germany to the U.S. for treatment, as was considered for AVR), requiring coordination of multilateral approvals. Training and emergency preparedness are also part of best practices – ensuring that transport personnel understand the unique nature of graphite waste (e.g., that if an accident occurred, responders should avoid inhaling any released graphite dust, which could carry ^{14}C and ^{60}Co). The industry also benefits from using proven commercial products, like ISO freight containers that meet IP-2 specifications, which simplifies regulatory approval and leverages common handling equipment.

8. VOLUME REDUCTION AND/OR SEPARATION

The decision to treat graphite to reduce volume or activity versus to manage it untreated is nuanced. The best industry practice is to perform an options assessment during decommissioning planning. Graphite's ability to encapsulate radionuclides (as evidenced by low leach rates) makes it a stable waste form on its own (Bucur, Ichim, & Florea, 2018). This argument prevailed in the U.S. DOE's approach – rather than oxidizing Fort St. Vrain's 2000 graphite blocks, DOE chose to package and store them intact, considering that deep disposal would eventually occur. The NEA and IAEA have highlighted research into removing the outer contaminated layer of graphite blocks, which can substantially lower the overall radionuclide inventory with minimal volume reduction. Such decontamination has been successfully demonstrated at small scale (e.g., grit blasting or chemical etching removing ~ 5 mm of graphite surface). If scaled up, this could allow large portions of graphite to be reclassified from ILW to LLW, suitable for cheaper near-surface disposal – a clear economic incentive in countries lacking a deep repository[14]. For instance, Russia has explored chemical treatment to decontaminate RBMK graphite, and China is reportedly considering graphite conditioning to meet near-surface disposal standards.

Potential Hazards in Volume Reduction Processes and Regulatory Implications

This section covers an analysis of potential hazards in various volume reduction processes being researched for reactor graphites with considerations for regulatory compliance. The incineration or other (chemical) volume reduction techniques proposed for graphite and management of the resulting ash are subject to various regulatory frameworks, particularly if the graphite has been used in industrial or nuclear applications. In the United States, regulations such as the Resource Conservation and Recovery Act (RCRA) govern the disposal of industrial waste, including ash from incineration processes (EPA, 2020). If the graphite was used in nuclear reactors, further regulations from the U.S. NRC apply, especially related to radioactive contamination (NRC, 2019).

Incineration is the simplest way to reduce the irradiated graphite waste volume into ash. Volume reduction is one major incentive for countries like France, Germany, and the U.K. to explore incineration of irradiated graphite, though it remains technically challenging to ensure adequate radiological safety during the volume reduction process. The estimated volume reduction ratios during incineration of various types of graphite, based on material properties and reported data in nuclear waste management literature are given in Table 12. The results are from studies in thermal oxidation (e.g., in TGA—thermogravimetric analysis—systems) that measured ash content. These laboratory values are then extrapolated to bulk waste forms in reactor decommissioning simulations.

[14] The UK does not consider that such surface treatment as a viable option. Contamination, especially that resulting from adventitious transport in the coolant circuit, becomes lodged inside the porous structure of the graphite because of differential pressures across the blocks driving a significant gas flow. This is especially true in AGR.

Table 12. Volume Reduction Efficiency Estimates by Incineration for Various Graphites

Graphite Type	Typical Initial Ash Content, %	Volume Reduction (Feed: Ash)[15]	Remarks
Natural graphite	> 5%	20:1 to 50:1	Contains mineral impurities like silicates and clays. Oxidizes but leaves more ash.
Electrode graphite	1 – 2 %	50:1 to 100:1	Synthetic graphite with fewer impurities. Burns efficiently with moderate ash content.
Nuclear graphite	< 0.5%	150:1 to 200:1	Ultra-pure graphite with minimal ash-forming impurities. Near total oxidation is possible under controlled conditions. European and Korean studies (e.g., EDF, Magnox) showed > 99% mass loss during controlled incineration (IAEA, 2024), (NDA, 2013), (Yang, 2004).
Irradiated nuclear graphite	< 0.5 %	100:1 to 200:1	Similar to nonirradiated nuclear graphite but requires gas treatment due to radioactive isotopes, for example ^{14}C, ^{3}H, and ^{36}C. Containment systems would be needed.

Volume reduction ratios were derived based on the oxidation behavior of graphite types, their ash content, and real-world incineration studies. The higher the ratio, the more efficient the volume reduction, which is crucial for minimizing long-term waste repository requirements.

Environmental Impact

The environmental impact of graphite incineration is generally low, particularly when the graphite is high purity. Oxidation produces primarily carbon dioxide (CO_2), which contributes to greenhouse gas emissions (IPPC, 2021). However, the ash itself, composed largely of inert minerals such as silica and alumina together with radioisotope compounds such as oxides and will constitute a very small quantity of intermediate-level waste (GTCC). (Pierson, 1993), (EPA, 2020).

There are several potential hazards that may be associated with the incineration of graphite. These are identified, along with common mitigation strategies and governing U.S. and IAEA regulations in Table 13.

[15] Actual ratios depend on incinerator design, feed form (bulk vs. shaped parts), and burn conditions (temp, O_2 availability). Radioactive graphite needs gas capture systems, and that can impact how much is burned in one batch.

Table 13. Hazard Analysis Considerations for Incinerating Irradiated Nuclear Graphite Waste

Hazard	Mitigation Strategies	Regulations (U.S. & IAEA)
Radiological release	HEPA and activated carbon filtration, real-time monitoring systems	10 CFR 61 & 40 CFR 190 (EPA) - Limits on radioactive emissions; IAEA Safety Standards GSG-9
Airborne contaminants	Advanced filtration, use of catalytic converters for NO_x/SO_x, controlled release rate	Clean Air Act (40 CFR 61 Subpart H); IAEA Safety Standards GSG-7; NRC 10 CFR 61, Waste Classification; NRC 10 CFR 20 Subpart K – Effluent Limits
Structural integrity of incineration system	Regular structural inspections, high-temperature-resistant materials	10 CFR 50 Appendix A (Safety Design); IAEA SSR- 2/1
Secondary waste generation	Solidification/stabilization of residual ash, dedicated disposal pathways	RCRA; IAEA NW-G-1.3
Public and worker radiation exposure	Remote handling, controlled access zones, improved shielding design	10 CFR 20 (Radiation Protection); IAEA GSR Part 3
Fire risk	Oxygen-controlled incineration, fire suppression systems, controlled airflows	NFPA 801 (Fire Protection for Facilities Handling Radioactive Materials); IAEA NS-G-1.7
Regulatory compliance challenges	Close regulatory alignment, real-time reporting to oversight bodies, engagement with public stakeholders	IAEA Safety Standards SSR-5; NRC Regulatory Guide 3.65

This survey of nuclear work examined whether irradiated graphite from nuclear reactors in the U.S. has been volume reduced via incineration, with a focus on potential efforts at Savannah River Site (SRS) or similar facilities. No large-scale incineration of irradiated graphite has been conducted at SRS or other U.S. sites.

Current U.S. policy has greater coverage of storage, characterization, and long-term disposal rather than thermal treatment. No international precedents have been found for irradiated graphite incineration (IAEA, 2006). Germany and the U.K. evaluated graphite incineration but ultimately chose disposal options. Globally incineration remains rare. When pursued, incineration would likely involve the following steps:

- Feed preparation – size reduction and sorting
- High-temperature combustion chamber (EPA 1999)
- Advanced off-gas system for ^{14}C and tritium capture
- Secondary waste treatment and regulatory air monitoring

Typically, high-temperature incineration (pressure) vessels would be flowing water-jacketed to maintain the structural integrity of the vessel to below the creep regime. However, water-based cooling systems present risks in nuclear graphite incineration due to potential leaks, hydrogen generation, and steam explosions. There have been cases of rupture of the incineration vessel due to water leak into the vessel resulting in steam explosions (Aria, 2017), (EPA, 1999).

Alternative cooling technologies offer enhanced safety and efficiency. Below are viable alternatives, their advantages, and potential challenges.

Air Cooling Systems: Air cooling is a simple and effective method that uses forced convection to dissipate heat. However, it is generally less efficient than liquid-based cooling and may not be suitable for high-temperature incinerators (IAEA, 2021).

Molten Salt Cooling: Molten salts, such as lithium fluoride-beryllium fluoride (FLiBe), can provide efficient cooling without the risk of steam explosions. They offer high thermal capacity and stability. However, molten-salt handling requires specialized corrosion-resistant materials (Qualls, 2018).

Liquid Metal Cooling: Liquid metals such as sodium, lead-bismuth, or gallium can be used for cooling in nuclear applications. They provide excellent thermal conductivity and reduce overheating risks. However, sodium is highly reactive with air and water, necessitating careful containment and lead is toxic, necessitating handling considerations. (NEA, 2015).

Gas-Based Cooling (Helium): Helium is usually the chosen circulating coolant in high-temperature nuclear reactors. Helium is chemically inert and does not react with graphite, ensuring safety. Sodium, potassium, and other alkali metals are often used as working fluid for high-temperature heat pipes because of their high boiling temperature, low saturation pressure, high latent heat of vaporization and high heat transfer performance. However, it requires high-pressure systems for effective heat transfer (IAEA, 2007).

Phase-Change Cooling (Heat Pipes): Heat pipes use phase-change fluids such as sodium or potassium to transfer heat passively; these alkali metals are often used as working fluid due to their high boiling temperature, low saturation pressure, high latent heat of vaporization and high heat transfer performance (Zhang, 2021). These systems are highly efficient and fail-safe as they rely on capillary action rather than pumps (Kingka Tech Industrial Limited, 2024).

To summarize, while water-based cooling is effective, alternative systems such as molten salts, liquid metals, and gas-based cooling provide enhanced safety and efficiency in nuclear graphite incineration. The selection of the appropriate cooling system depends on operational temperatures, material compatibility, and safety considerations.

Studies of graphite incineration in industrial and research settings support the conclusion that the resultant ash production is minimal. For example, graphite electrodes used in steelmaking processes are regularly

incinerated in controlled furnaces, and the residual ash content is consistently below 1%, aligning with laboratory measurements (Marsh & Rodríguez-Reinoso, 2006). Similarly, in nuclear decommissioning projects, incineration of graphite from reactor cores has been extensively studied, with results showing ash production directly correlating to initial impurity levels (IAEA, 2006).

Use of Rumsfeld's Risk Assessment Decision Making

The technical and societal risks, primarily associated with community involvement and proper communication, associated with irradiated graphite incineration can also be analyzed using Rumsfeld's decision-making matrix. A summary of this presented in Table 14

Table 14. A Rumsfeld Decision-Matrix Risk Analysis Associated with Irradiated Graphite Incineration

Category	Technical Considerations	Societal Considerations
Known knowns	Emission control technologies, filtration systems, shielding	Regulatory requirements and safety standards
Known unknowns	Long-term effects of incineration on waste composition	Public perception of nuclear waste incineration
Unknown knowns	Potential new filtration materials not yet tested in nuclear settings	Unaccounted economic impacts on waste management programs
Unknown unknowns	Undiscovered material reactions at high temperatures	Future societal resistance due to unforeseen risks

Based on NRC's NUREG/KM-0016 (Gavrilas, 2021), current regulatory practices are sufficient to assure reasonable safety for risks associated with "known knowns". Long-term effects of incineration on waste composition are a difficult technical challenge to address, though there may be scenarios in which it could be used with minimal impact on risk. Technical efforts related to "unknown knowns" and "unknown unknowns" are likely not justified to address based on economics until they become fully known and accepted to impact risk significantly.

9. TECHNOLOGY LEVELS AND REGULATORY IMPLICATIONS OF GRAPHITE WASTE TREATMENT METHODS

For this work, the Technology Levels of various graphite waste treatment technologies were considered in order to identify any associated regulatory challenges and implications for the U.S. NRC. Prior findings on the technical maturity of volume reduction techniques, along with a detailed analysis of the limitations in using nonirradiated materials and early-stage experiments in regulatory decision-making are provided below.

The DOE uses 9-level criteria (DOE, 2020) to assess the technical readiness levels of emerging technologies. These levels progress from basic scientific observation to full commercial deployment. This work does not make TRL assessments but examines the challenges that could be faced by the regulatory staff in assessing these methods from a regulatory perspective.

Lack of Beyond Bench-Scale and Nonirradiated Data

Many graphite treatment methods have only been tested on nonirradiated graphite or surrogate materials, raising concerns about extrapolating to real irradiated graphite. Such materials differ significantly in microstructure, radionuclide content, and chemical reactivity. These differences affect oxidation behavior, leachability, and overall treatment effectiveness. Engineering scale-up challenges include dust management, off-gas containment, and secondary waste treatment (Fuks et al, 2020), (Gallego, Burchell, & Srinivasan, 2009). A list of various treatment methods is provided in Table 15 alongside a discussion of current technical readiness and resultant regulatory implications.

Table 15. Technology Levels and Regulatory Implications of Graphite Waste Treatment Methods

Treatment Method	Technical Summary	Regulatory Implications (NRC)
Incineration / oxidation	Proven in pilot-scale for nonirradiated graphite (France,); off-gas capture demonstrated; full- scale on irradiated graphite not yet done; significant engineering and radiological barriers remain (Goodwin & Bradbury, 2013), (IAEA, 2016).	Yet to have demonstration with irradiated graphite. May be acceptable in principle, but approval, in part, hinges on performance validation with irradiated graphite and emissions control.
Thermal treatment (annealing / plasma)	Lab-tested for tritium removal and surface oxidation. Ineffective for ^{14}C. Plasma surface cleaning shows promise but only tested on small samples.	May support waste downgrading if shown effective. This would likely be treated as supplemental/surface treatment unless full decontamination is proven.

Chemical leaching / electrochemical	Bench-scale only. Ineffective for ^{14}C, limited to surface contamination. Lab-only studies on nonirradiated graphite. Electrokinetic methods are still experimental.	Likely insufficient maturity for licensing as standalone. Incorporation as part of a multi-stage treatment could be useful. Proof of radionuclide removal from irradiated graphite is lacking.
Microbial degradation	Conceptual phase. Few lab studies (e.g., PNNL, PBMR South Africa). No engineered system or pilot data. Targeting ^{14}C via slow biological oxidation.	Likely far from deployable. Currently a long-term R&D topic. There is a lack of evidence to allow for licensing decisions in the current state.
Recycling and reuse	Proposed for isotope recovery (e.g., ^{14}C) or shielding use. Technical and economic barriers are high. Limited bench-scale studies. No demonstrated purification pathway exists.	Likely would require radionuclide inventory traceability, radiopurity standards, and health risk analysis. No licensing basis without scale-up and real-waste trials.

NRC Regulatory Decision-Making Needs

The NRC requires high-confidence data for licensing decisions, especially when classifying or reclassifying radioactive waste. Bench-scale studies may inform research, but without testing on irradiated materials and demonstrations at scale, the NRC is unlikely to credit these methods with waste hazard reduction. Many regulatory decisions have been based on bounding analyses, pilot testing, and robust QA programs. In the absence of industrial validation, graphite waste remains classified for deep geological disposal or long-term storage (GAO, 2020), (Tracyhonney, 2024).

NRC Evaluation of Technologies with Uncertainty and Low Maturity

Although the NRC does not use TRLs, its licensing philosophy aligns with the need for mature, demonstrated technology. Licensing evaluation typically requires:

- Conservative assumptions and bounding analyses
- Repeatable, peer-reviewed data
- Demonstration in relevant operational environments
- QA-controlled test procedures
- Stepwise approvals (e.g., pilot-to-full scale transitions)

Technologies that are not mature are generally not accepted for routine licensing decisions when the technology and its use are safety significant, particularly for high-stakes waste streams like irradiated graphite.

10. ASSESSMENT OF U.S. NRC REGULATIONS FOR TECHNICAL NEUTRALITY IN MANAGING IRRADIATED GRAPHITE

This section assesses the technical neutrality of NRC regulations, as they apply to the conditioning, treatment, packaging, transport, and storage of irradiated graphite waste[16] from nuclear reactors. Technical neutrality refers to a regulatory approach that is performance-based and does not prescribe specific technologies, allowing flexibility while ensuring safety and compliance.

Packaging and Transportation

The NRC regulates the packaging and transportation of radioactive materials, including irradiated graphite, under 10 CFR Part 71. These regulations ensure that packages used for transporting irradiated graphite meet stringent containment, shielding, and heat dissipation criteria. The rules are technically neutral, focusing on ensuring that the total radiation dose to workers and the public remains within regulatory limits, regardless of the specific type of radioactive material or its physical form. The transportation of spent nuclear fuel has been conducted safely over several decades. Since 1965, approximately 3,000 shipments have been completed in the United States without any harmful release of radioactive material. This exemplary safety record underscores the effectiveness of the regulatory framework and the robustness of the transportation casks used (Adkins, 2006).

Treatment and Conditioning

The NRC regulations shown in Table 2 of this report do not prescribe specific technologies for the treatment or conditioning of irradiated graphite. Instead, the NRC adopts a performance-based approach, which requires licensees to demonstrate any treatment process (e.g., oxidation, encapsulation, or leaching) effectively controlling the release of radionuclides, thermal stability, and structural integrity of the container. These evaluations must show that the treated graphite meets WAC for subsequent storage or disposal. This regulatory flexibility supports the development of innovative treatment technologies for irradiated graphite, including thermal and chemical processes.

Storage

The interim storage of irradiated graphite is regulated under 10 CFR Part 72, which governs the independent storage of GTCC, spent nuclear fuel and high-level radioactive waste. Storage facilities are required to demonstrate that systems and components are designed to prevent the release of radioactivity and protect public health. NRC regulations allow for site-specific licensing and do not advocate any specific storage designs or technologies, ensuring technical neutrality.

Summary of Technical Neutrality of NRC Regulations

The relevant NRC regulations emphasize performance-based criteria rather than design prescriptions. This regulatory philosophy allows for the use of diverse and innovative methods to handle irradiated graphite provided safety outcomes are achieved. For example, licensees may propose methods such as incineration, plasma oxidation, or encapsulation for irradiated graphite, provided they can demonstrate adequate control of radionuclide release (e.g., ^{14}C, tritium, or ^{36}Cl), and compliance with dose and containment limits.

---

[16] Irradiated graphite is taken here and henceforth taken as a synonym for the group of carbonaceous waste, as long as no other relevance for other carbonaceous waste forms are directly addressed.

This neutrality supports flexibility, innovation, and adaptation of common international practices. Because irradiated graphite contains long-lived isotopes and has different properties from nonirradiated materials, licensees need to submit comprehensive data and analysis to support any proposed treatment or storage method.

NRC's regulatory framework for conditioning, treating, packaging, transporting, and storing reactor-irradiated graphite waste is technically neutral. By relying on performance objectives instead of prescriptive rules, the NRC enables the safe use of emerging technologies tailored to the unique challenges of irradiated graphite. The licensing is based on the designed treatment performance and containment under realistic conditions, which should consider the complex radiological and physical nature of irradiated graphite.

11. TECHNICAL CHALLENGES FOR REGULATORS (NRC) AND WASTE MANAGERS

The safe management of irradiated graphite has revealed several technical challenges that should be addressed by regulators and waste management organizations.

Cask Design and Storage System Evaluation

Graphite waste packages might need custom designs – for example, a cask that can hold many graphite blocks or drums. The NRC will likely verify that such a design meets normal and hypothetical accident conditions: mechanical integrity, no excessive loss of shielding (graphite's radioactivity is lower, but shielding against any ^{60}Co gammas must remain), and containment of particulates. Graphite blocks can move and scrape to create dust, which is electrically conductive and could foul seals or filters. Ensuring the cask's robust containment boundary against fine particles is a specific concern. Staff also evaluate thermal performance; while decay heat in graphite waste is low, in an accident fire (per 10 CFR 71 testing), a large graphite mass might absorb heat and potentially oxidize if oxygen enters. Cask designs may thus incorporate an inert (argon) filling to prevent oxidation in the event of seal failure during a fire. NRC reviewers should be cognizant of such failure modes and may request analysis of possible oxidation-associated hazards inside a transport cask in severe scenarios. This is a non-standard issue, as spent fuel casks have inert filler gas but are not concerned with the content oxidizing. The cask designer might need to demonstrate that even if some graphite is oxidized, the cask would contain the gaseous oxidation products.

Waste Form Qualification and Acceptance Criteria

When disposing of graphite waste, especially in near-surface facilities, waste form criteria must be met to ensure long-term stability. One technical challenge is demonstrating that graphite disposal will not lead to unanticipated environmental releases. For shallow land disposal, intruder scenarios are evaluated, such as what if someone drills into it in a few hundred years? Graphite blocks are physically stable, but an intruder could, for instance, use them as building material or inhale dust. Thus, waste forms processing may involve pre-treating the graphite (e.g., immobilizing it in grout) to make it essentially immovable and benign to an intruder. NRC staff, in reviewing a disposal facility's license or a site's Performance Assessment, may assess the dose to potential intruders from long-lived isotopes like ^{14}C in graphite. ^{14}C can be incorporated into CO_2, which could be inhaled or into plants if it enters soil gas. This makes modeling of ^{14}C in the environment critical. ^{14}C is a mobile, long-lived radionuclide, and its behavior, whether it stays as a solid carbonate or becomes a gas, depends on chemical form, and can result in it becoming mobile. Various organizations have invested in research such as the CAST project[17] and DOE's performance assessments[14] to better predict ^{14}C release. A finding from these studies is that only a small fraction of ^{14}C will leach or gasify under likely conditions (Bucur, Ichim, & Florea, 2018), which supports compliance with dose criteria in many scenarios. However, regulators often apply conservative assumptions such as considering the possibility that a certain percentage of ^{14}C will become methane and reach the surface. Ensuring dose compliance (typically less than 0.25 mSv/yr for the public in NRC frameworks for disposal) over millennia is a non-trivial analytical task and an ongoing technical challenge for graphite waste disposal.

---

[17] The CArbon-14 Source Term (CAST) project was a European initiative aimed at understanding the release mechanisms of ^{14}C from radioactive waste materials under conditions relevant to geological disposal facilities. The project focused on the release of ^{14}C as dissolved and gaseous species from irradiated metals (steels, zircalloys), irradiated graphite, and ion-exchange materials.

Environmental Impact Assessments and Safety Considerations

Both NRC and DOE (and their international counterparts) will evaluate the environmental impact of any long-term graphite waste management solution. This involves accounting for very long-time frames and uncertainties in geochemical behavior. DOE's GTCC EIS, for instance, had to consider the impacts of disposing of graphite in different geologies. One consideration was ^{14}C's contribution to groundwater or atmosphere over time – even though the release is slow. ^{14}C is biologically relevant: when it is present as carbon dioxide, it can be taken up by plants and enter the food chain. Environmental modeling should include ^{14}C transport in geosphere and biosphere. Understanding and predicting this behavior is currently based on data from experiments like those in the U.K. and EU where graphite was oxidized at low temperature to obtain information on the forms of ^{14}C release. One result was that under anaerobic (reducing) repository conditions, ^{14}C may be released as hydrogen containing organic gases (like methane) which migrate differently than CO_2 (Bucur, Ichim, & Florea, 2018).

The environmental assessment also covers operational impacts: for example, if a treatment plant is built to incinerate graphite, what are the emissions and risks to the public? A case study is the proposed graphitic matrix digestion at SRS to treat German pebble fuel graphite (McWilliams, 2015). That process used nitric acid at high temperatures to oxidize graphite. Using nitric acid at elevated temperatures for graphite oxidation entails safety risks involving thermal, chemical, and radiological hazards. Nitric acid reacts exothermically with graphite. Elevated temperatures accelerate this reaction, increasing the risk of thermal runaway events (Venugopal & Prasad, 1983). During oxidation, toxic nitrogen oxides (NO and NO_2) are released. These gases are hazardous to personnel and corrosive to equipment (Evenson, 2002). High-temperature nitric acid is corrosive to construction materials, requiring special materials of construction to prevent system failure (DOE, 1992). Any organic matter contamination in irradiated graphite and graphitic matrix material could cause violent or explosive reactions with nitric acid under heat (Urben, P.G. (Ed), 2006). Thermal oxidation of irradiated graphite in nitric acid can release ^{14}C and ^{3}H into the gas phase (IAEA, 2006).

Gas evolution (CO, CO_2, NO_x) may lead to over pressurization and vessel rupture if not properly vented (OECD/NEA, 2016). Residues from nitric acid oxidation of graphite may be pyrophoric, especially when dried and exposed to air (Stevens & Marsh, 1987).

Safe application requires comprehensive risk mitigation, including temperature control, adequate ventilation, corrosion-resistant materials, and radiological monitoring. NRC does not regulate DOE's R&D, but any similar process for commercial waste would undergo rigorous environmental and safety review (air permits for ^{14}C release, etc.).

Proof-of-concept experiments have also been performed and patented by using nonirradiated graphite and liquid-phase oxidative digestion methods to facilitate the removal of elements attached to carbonaceous material (Pang, 2021). There are several other studies that are of basic research nature, and the results do not yet provide sufficient information to assess hazards associated with such processes and to examine the applicability of appropriate NRC regulations and guidance to such emerging studies.

Radionuclide Content, Dose, and Environmental Impact

The radionuclide content of irradiated graphite – chiefly ^{14}C, ^{36}Cl, ^{3}H, and ^{60}Co – drives the radiological safety considerations. Dose compliance is assessed for several scenarios: workers handling the waste, the public during transport, the public during storage/disposal (including post-closure of a repository), and intruders or environmental migration long after disposal.

For workers and operational scenarios, compliance is maintained by engineering controls (shielding, remote handling) and administrative controls (time, distance, shielding, contamination control training). Graphite's dose rate can vary widely; freshly discharged graphite with high ^{60}Co may have contact dose rates in the mSv/hr range, requiring shielding and remote tools. But after decades, dose rates drop significantly. Experience at BGRR and other projects has shown that with simple precautions like containment tents and filtered ventilation, worker doses can be kept ALARA (DOE, 2010). Because graphite is mostly a beta emitter, internal dose from ingestion/inhalation of graphite and graphitic matrix dust is a concern – requiring good dust suppression. Strict adherence to radiation work permits and protective equipment ensured compliance with dose limits in those case studies (for example, BGRR reported successful removal of the graphite with no spread of contamination and doses below project targets (Kneitel, 2014)).

For the public and environment, the long-term phase is critical. ^{14}C is a major contributor to potential doses due to its long half-life and ability to become CO_2 (which could be inhaled or absorbed by plants). Environmental impact assessments, like DOE's GTCC EIS[18] and various international performance assessments, generally concluded that disposing of graphite in a well-engineered facility results in doses to the public that are a small fraction of regulatory limits (GAO, 2022). Engineered barriers (concrete, clay, metal liners) and the low release rate from graphite combine to isolate the radionuclides. For example, the U.K. estimated that for one studied scenario, even under worst-case assumptions, ^{14}C release from a geological disposal facility with graphite waste would lead to doses on the order of 1E-3 to 1E-2 mSv/yr, well below the guidance level of ~ 0.3 mSv/yr for such facilities (Grambow, et al., 2013). Nevertheless, sensitivity analyses are ongoing. One environmental concern is the speciation of ^{14}C: as noted, under anaerobic conditions ^{14}C can form organic gases (like methane), which might travel through rock faster than dissolved carbonate. If ^{14}C releases as methane and reaches the biosphere, it oxidizes to $^{14}CO_2$ and can be incorporated into agricultural products, which is an exposure pathway to people. Countries like France and the U.K. are conducting research to better quantify this behavior (NDA, 2016), (Limer, 2017). Additionally, ^{36}Cl, while usually in lower activity, is very mobile in water if it is in the carbonate form and can contaminate the environment via water pathways. Ensuring that the total inventory of ^{36}Cl does not threaten groundwater could involve controlling the concentration or mixing waste types to avoid a concentrated source of ^{36}Cl) (NRC, 2000).

Dose compliance during normal transportation is achieved by following DOT radiation limits for packages and vehicles. Typically, doses to the public from routine shipments are extremely low – the packages might give off at most a few microsieverts per hour at contact. Past shipments of irradiated graphite have not reported any incidents of excessive radiation; they are comparable to or lower than spent fuel shipments in terms of dose to the public (which studies have shown are negligible, on the order of microsieverts per trip)[19].

Environmental impact during interim storage could be designed to be negligible. The storage owner should ensure that any water) that contacts the graphite is contained and treated, to prevent release of tritium or other soluble species. For example, if graphite is stored in a facility with a roof, one should plan how to handle water in case of sprinkler activation, as water can leach tritium out. A facility without a roof may need to

[18] In the United States, the Department of Energy (DOE) conducts performance assessments to evaluate potential releases of radioactivity, including ^{14}C, from disposal facilities into the environment. These assessments are quantitative evaluations that consider various factors such as the performance of engineered barrier systems, release and migration of radionuclides, and resultant radiological doses to human receptors.

[19] The transportation of spent nuclear fuel has been conducted safely over several decades. Since 1965, approximately 3,000 shipments have been completed in the United States without any harmful release of radioactive material. This exemplary safety record underscores the effectiveness of the regulatory framework and the robustness of the transportation casks used (Adkins, 2006).

consider water from rain.

In essence, compliance with dose and environmental protection criteria for irradiated graphite waste is achievable with existing methods, but with careful analysis and conservative planning.

Irradiated nuclear graphite (i-graphite) waste management encompasses various technologies and is subject to regulatory frameworks established by organizations such as the IAEA and the U.S. NRC.

Long-Term Integrity of Storage

As interim storage periods lengthen (potentially many decades or even centuries), technical challenges arise in ensuring that storage systems remain safe. To comply with defense-in-depth requirements, graphite waste packages could need periodic surveillance for any degradation. Metal containers could corrode, or concrete silos could crack. NRC and DOE guidance for extended storage of LLW requires monitoring systems and aging management. One specific issue is graphite's tendency to absorb moisture and then release ^{14}C or ^{3}H as gases. If storage is in a humid environment, graphite might gradually pick up water which can produce tritiated water or ^{14}C-labeled CO_2. While the production of these molecules would be gradual, over a long period in a sealed building these molecules could accumulate. The technical challenge is predicting issues that might occur in 50+ years with a material that has not often been stored that long outside a reactor. Options for extracting accumulated water should be taken into account.

Cross-Agency Coordination

Carbonaceous waste management involves regulation from multiple agencies. When physical transfer occurs between different locations, this can involve changes in which agencies will be regulating the waste. NRC oversees commercial waste generation, DOT oversees transport, DOE may take possession for disposal, EPA may have a say in environmental standards, and state regulators may oversee disposal sites. A challenge for technical experts is ensuring that all regulations harmonize. For example, if a commercial entity wanted to dispose of irradiated graphite at DOE's WIPP – which is licensed for TRU waste – it would require regulatory accommodations since WIPP is not for commercial waste and graphite is not a TRU waste by the NRC definition. These interagency issues are usually resolved at policy levels, but technical staff may contribute to analysis and evaluation. The 2022 GAO report on GTCC waste highlights that without clear disposition paths, some LLW might lack an identifiable planned disposal site (GAO, 2022). Unknown aspects of waste management create uncertainties that complicate decommissioning planning and technical decision-making.

12. SUMMARY

Irradiated graphite presents a multifaceted challenge at the intersection of nuclear engineering, material science, and regulatory policy. This technical report has summarized an analysis and evaluation of the current state of knowledge and practice in regulating, treating, transporting, and storing irradiated graphite waste.

Key Conclusions

A comprehensive review of the NRC regulations, along with relevant regulatory guidance documents, reveals that the NRC framework is both performance-based and technically neutral. This regulatory approach does not prescribe specific technologies but instead sets safety and environmental performance objectives that licensees must meet. Such flexibility enables license holders to explore and implement innovative methods for managing irradiated graphite waste—such as incineration, encapsulation, or advanced conditioning techniques—without requiring regulatory changes, as long as it can be demonstrated that the chosen method meets established safety, health, and environmental standards.

Importantly, while the NRC permits technological innovation, it still requires: (a) robust safety demonstrations, including modeling of radiological hazards and containment integrity, and (b) thorough environmental assessments, aligned with the National Environmental Policy Act (NEPA) and NRC's own licensing procedures.

This balance of flexibility and rigor positions the NRC framework as adaptable to evolving technologies while maintaining high standards for public and environmental protection.

Regulatory Alignment is Crucial

A clear understanding of NRC, IAEA, DOE, and DOT regulations is foundational. Presently, U.S. irradiated graphite from commercial sources is likely to be treated as GTCC waste and is pending a disposal solution, with DOE partially responsible for interim management. Internationally, approaches range from near-surface disposal under strict criteria to indefinite safe storage for lack of options. Harmonizing these approaches with safety standards ensures that despite different strategies, protection of human health and environment remains paramount. Future policy developments, such as DOE’s decisions and ultimate implementations on GTCC disposal, as well as international disposal collaborations, will significantly influence graphite waste management pathways.

Technical Feasibility of Safe Management

Advanced modeling and experimental studies reinforce that irradiated graphite can be managed safely. Graphite has the ability to retain radionuclides like ^{14}C with minimal release. Therefore, even untreated graphite can be a stable waste form in a suitable repository. At the same time, industry experience and R&D have yielded effective treatment technologies (thermal, chemical, mechanical) that can reduce waste volumes and isolate radionuclides when needed.

No insurmountable technical barrier has been identified; rather, the choice of direct disposal vs. treatment is an optimization problem balancing safety, cost, and regulatory acceptance.

Industry Practices and Innovation

Industry practices for volume reduction (such as incineration with off-gas capture, or surface decontamination) and for safe transportation/storage (using LSA packaging and robust interim stores) have

been demonstrated in case studies. Projects like the removal of 60,000 graphite blocks from BGRR to a disposal site showed that large-scale graphite waste operations can be conducted efficiently and safely. The industry continues to innovate – from developing robotic dismantlement tools to exploring biotechnological treatments – which will make future projects even more streamlined. A notable practice is stakeholder engagement because graphite reactors were often early-generation reactors, their decommissioning is closely observed by the public. Transparent communication of how graphite waste is being handled (as done at sites like Windscale and BGRR) is helpful towards maintaining public trust.

Challenges for NRC and Technical Staff

Ensuring cask designs for any graphite transport are robust against oxidation and contamination release, verifying that long-term storage of graphite will not degrade containment, and reviewing performance assessments for any disposal facility involving graphite is paramount. These challenges are being met with ongoing research and updated guidance. For instance, insights from international projects on ^{14}C release are feeding into NRC's knowledge base.

Radionuclide and Environmental Focus

The long-term radiological risk of irradiated graphite is dominated by a few isotopes, chiefly ^{14}C, which necessitates a conservative approach to disposal. Yet, studies consistently show that with engineered barriers and the inherent stability of graphite, even worst-case scenarios yield doses within regulatory limits. EIS have validated deep geological disposal as a safe end state for graphite waste, and they highlight the importance of containing ^{14}C to prevent its migration into the biosphere. Thus, managing graphite waste is less about mitigating immediate high radiation (as in spent fuel) and more about ensuring containment over very long periods. The objective then shifts the focus to materials longevity, geochemical stability, and institutional controls, all of which are areas ripe for further work.

Outlook – Toward Sustainable Solutions

The future of irradiated graphite waste management is likely to be shaped by enhanced international cooperation and scientific advancement. Continued research into radionuclide removal (e.g., extracting ^{14}C) could eventually allow the reclassification of much graphite waste to lower categories, easing disposal. If advanced reactor designs incorporate lessons learned, the next generation of graphite moderators will be easier to dispose of or recycle, breaking the current waste management planning impasse for irradiated graphite. In the interim, the nuclear community recognizes that safe storage can be maintained for many decades, delaying implementation of permanent solutions without compromising safety. The coming years may see pilot implementations of waste management for small high-activity graphite components.

In conclusion, the management of irradiated graphite does not have any insurmountable technical or regulatory barriers, but deployment of future technologies relies on focused research and experimental validation, planning, rigorous technical analysis, and adherence to regulatory frameworks. By leveraging the extensive knowledge gained from modeling studies, experiments, and decommissioning projects to date, nuclear operators and regulators can ensure that irradiated graphite is handled in a manner that protects current and future generations. With continued innovation and collaboration, the remaining challenges can be overcome, enabling safe management of wastes from legacy graphite reactors and future nuclear reactors.

REFERENCES

Adkins, e. (2006, October). Spent Fuel Transportation Package Response to the Baltimore Tunnel Fire Scenario. *NUREG/CR-6886, Revision 2*. Richland, Washington, U.S.A.: Pacific Northwest National Laboratory.

Andra. (2015). *Graphite Reference Guide - State of Knowledge* (Vols. Andra 398-B). Châtenay-Malabry cedex, France: Andra - French National Radioactive Waste Management Agency.

Andrews, & et.al. (2005). Evaluation of the Potential for Carbonaceous Dust Explosibility in the Decommissioning of the Windscale Piles. *Proc. CESEP'05, 1st International Conference on 'Carbon for Energy Storage and Environment Protection'.* Orléans, France.

ANSTO. (2025). ANSTO Synroc® Treatment Facility. Retrieved March 19, 2025, from https://www.ansto.gov.au/products/ansto-synroc/facility

Aria. (2017, April 14). Explosion in the feed hopper of a waste incinerator. *E38.21 - Treatment and disposal of non-hazardous waste*.

Arm et al, 2. (2022, June). Plan for Developing TRISO Fuel Processing Technologies. *PNNL-32969*. Pacific Northwest National Laboratory.

ASTM. (2005). Standard Specification for Isotropic and Near-isotropic Nuclear Graphites. West Conshohocken, PA, U.S.A.: ASTM International.

Austin, J. (1994, November 29). OECD/NEA Questionnaire on radioactive waste legislation and policies. *NRC Memorandum*. Washington, D.C., U.S.A.: USNRC. Retrieved May 25, 2025, from https://www.nrc.gov/docs/ML0318/ML031810741.pdf

Beer, H.-F. (2009). Safe and inexpensive. Complete conditioning of activated reactor graphite,. *Strahlenschutz Praxis (Koeln); Journal, 15*(4).

Bespala, e. (2018). About chemical form and binding energy. *Nuclear Energy and Technology 4(1): 51–56, 4*(1), 51-56. doi:https://doi.org/10.3897/nucet.4.29855

BfS. (2019). *German Nuclear Waste Management Strategy and Practices. Report.* Bundesamt für Strahlenschutz.

Bisplinghoff, M., Moormann, R., Nabbi, R., & Kugeler, K. (2005). Radiochemical characterisation of graphite from Jülich experimental reactor (AVR). *IAEA, Paper 13*. International Atomic Energy Agency. Retrieved March 31, 2025, from https://www-pub.iaea.org/MTCD/publications/PDF/ngwm-cd/PDF-Files/paper%2013%20%28Bislinghoff%29.pdf

Blackwood, D. J., Otlet, R. L., & and MacKenzie, A. B. (1992). Leaching studies on irradiated reactor graphite to estimate potential radiocarbon releases to the environment. *Waste Management, 12(2-3)*, 149-158.

BNES (British Nuclear Energy Society) 1987, 'Chernobyl - A Technical Appraisal) ISBN 0 7277 0394 3

Bradbury D. and Goodwin J.C. 2010, 'Graphite Reuse & Recycle'; CARBOWASTE Technical Report (T-

5.0.0); https://igdtp.eu/activity/carbowaste/]

Brookhaven National Laboratory (BNL). (1997). Final Environmental Impact Statement for Decommissioning of the Brookhaven Graphite Research Reactor. *DOE/EIS-0256*. U.S. Department of Energy (DOE).

Bucur, C., Ichim, C., & Florea, I. (2018). 14C Release from TRIGA Irradiated Graphite. *Radio Carbon, 60*(6), 1819-1829. doi:https://doi.org/10.1017/RDC.2018.131

Burchell, T., & Pappano, P. (2010). *The Characterization of Grade PCEA Recycle Graphite Pilot Scale Billets.* Oak Ridge, TN, U.S.A.: Oak Ridge National Laboratory.

Burchell T. D and Pappano P. J., 2010, 'Recycling Irradiated Nuclear Graphite – A Greener Path Forward'; Proceedings of HTR 2010; Paper 068.

CORDIS - EU. (2024, 06 18). Final Report Summary - CARBOWASTE (Treatment and Disposal of Irradiated Graphite and Other Carbonaceous Waste). *Treatment and Disposal of Irradiated Graphite and Other Carbonaceous Waste*.

Cox, E. J., Harker, J., Wickham, S. M., & Noy, D. J. (2011). Development of a model to predict the 14C inventory in irradiated graphite. *Nuclear Instruments and Methods in Physics Research Section B: Beam Interactions with Materials and Atoms, 269, 11*, 2762-2773.

DeGrassi, G., Nie, J., & Hofmayer, C. (2008). Seismic analysis of large-scale piping systems for the JNES-NUPEC ultimate strength piping test program. *NUREG/CR-6983*. Washington, DC, U.S.A.: U.S. Nuclear Regulatory Commission.

DeLeon, G., & Maxted, M. (2014, July 10). Potential Acceptance and Disposition of German Pebble Bed Research Reactor Highly Enriched Uranium (HEU) Fuel Environmental Assessment. *Presentation to the Governor's Nuclear Advisory Council*.

Deng, J., Wu, Z., Xie, X., & Li, X. (2008, September 15). Treatment and disposal of waste graphite from nuclear facility decommissioning. *Nuclear Safety*, 49-51. Retrieved from https://www.osti.gov/etdeweb/biblio/22512759

Dietrich, G., THTR fuel handling and data, Slides, 2024

DOE. (1992). Chemical Hazards. *Nuclear Facility Training Handbook, Rev. 0, Module 3, DOE-HDBK-1015-3-92*. Washington, DC, U.S.A.: U.S. Department of Energy.

DOE. (2010, October). Brookhaven Graphite Research Reactor Graphite Pile Removal. *Final Closeout Report*.

DOE. (2020). Technology Readiness Assessment Guide 2020. *DOE G 413.3-4A*. U.S. Department of Energy,.

DOE. (2021). Radioactive Waste Management Manual. *DOE M 435.1-1*. U.s. Department of Energy. Retrieved March 28, 2025, from https://www.directives.doe.gov/directives-documents/400-series/0435.1-DManual-1-chg3-ltdchg-1/@@images/file

DOE, U. (2016, January). Final Environmental Impact Statement for the Disposal of Greater Than Class (GTCC) Low Level Radioactive Waste and GTCC-Like Waste (DOE/EIS-0375).

Dunzik-Gougar, M; Williams, P; Mafishane, I; Ashy,G. CARBOWASTE. 2012

Duwe, R. 1998, 'Freisetzung Gasförmiger Radionuklide aus HTR-Brennelementen unter Zwischenlagerbedingungen', Interner Bericht ZFK/HZ-IB-1/98

Duwe R. and Pott G., Storage Behavior of AVR Fuel Elements. Proc 3rd Int Seminar on Radioactive Waste Products (Radwap), Würzburg, Report Energy Technology, Vol 2, Research Center Jülich (1997) 399–402

Ebadat, E. (2009, May). Dust Explosion Hazard Assessment Including OSHA Combustible Dust NEP". *Annual OSHA & Workshop Safety Conference,*.

Enresa. (2020). *National Radioactive Waste Management Plan.* 7th Edition.

EPA. (1999, June). Report on Emergency Incidents at Hazardous Waste Combustion Facilities. *EPA530-R-99-014*. U.S. Environmental Protection Agency. Retrieved March 20, 2025, from https://archive.epa.gov/epawaste/hazard/tsd/td/web/pdf/increp5.pdf

EPA. (2020). *Resource Conservation and Recovery Act (RCRA)*. U.S. Environmental Protection Agency.

Evenson, e. a. (2002). Nitric Acid Oxidation of Graphite: Gas Evolution and Surface Chemistry. *Carbon, 40, 6*, 911-917.

Fachinger. (2016, May). Impermeable graphite; A new development for embedding radioactive waste. *Processing of Irradiated Graphite to Meet Acceptance Criteria for Waste Disposal. Results of a Coordinated Research Project. Companion CD-ROM*. International Atomic Energy Agency.

Fanny, G. (2013). Long term management of graphite waste in France - 13106. *International Conference and Exhibition on Decommissioning Challenges - Industrial Reality and Prospects.* Avignon, France.

Federal Register. (2020, January 23). Requirements for the Storage of Spent Nuclear Fuel. *A Proposed Rule by the Nuclear Regulatory Commission*.

Fuks et al, 2. (2020). Management of Radioactive Waste Containing Graphite. *Energies, 12, 4638*.

Gainey, B. (1976). *A Review of Tritium Behavior in HTGR Systems.* GA-A13461, General Atomic.

Gallego, N. C., Burchell, T. D., & Srinivasan, M. (2009). Milestone Report on the "Workshop on Nuclear Graphite Research". *ORNL/NRC/LTR-09/03*. Oak Ridge National Laboratory.

Gallego, N., & Burchell, T. (2011, September). A Review of Stored Energy Release of Irradiated Graphite. *ORNL/TM-2011/378*. Oak Ridge, TN, U.S.A.: Oak Ridge National Laboratory.

Galy, e. a. (2016). *Behavior of 14C in irradiated nuclear graphite waste: consequences for inventory, decontamination and disposal.* OECD/NEA. Retrieved from https://www.oecd-nea.org/rwm/wpdd/predec2016/docs/P-3-3___abs__GALY__.pdf?utm_source=chatgpt.com

GAO. (2020, February 11). Technology Readiness Assessment Guide: Best Practices for Evaluating the Readiness of Technology for Use in Acquisition Programs and Projects. *GAO-20-48G*. U.S. Government Accountability Office.

GAO. (2022, September). Nuclear Waste: DOE Needs to Improve Transparency in Planning for Disposal of Certain Low-Level Waste. *GAO-22-105636*. Washington, DC, U.S.A.: U.S. Government Accountability Office.

Garankin, J., Plukienė, R., Juodis, L., Duškesas, G., Lingis, D., Abdulajev, V., & Plukis, A. (2018). About rapid analysis method for the determination of 14C specific activity in irradiated graphite. *Nuclear Energy and Technology, 4*(1), 51-56. doi:https://doi.org/10.3897/nucet.4.29855

Gavrilas, e. a. (2021, March). Be riskSMART: Guidance for Integrating Risk Insights into NRC Decisions. *NUREG/KM-0016*. U.S. Nuclear Regulatory Commission.

General Atomics. (n.d.). *TRIGA Nuclear Reactors*. Retrieved from https://www.ga.com/triga/: https://www.ga.com/triga/

Goodwin, J., & Bradbury, D. (2013, January 31). Recycled Products for Nuclear Reactor Applications. *Technical Report (D-5.4.3)*. EURATOM.

Grambow, B., S., N., Petit, L., Blin, V., J., C., & de Visser, E. (2013, April 19). Disposal behavior of irradiated graphite and carbonaceous waste (Final Report). *CARBOWASTE - Seventh Framework Programme, Work Package 6.*

Graphitech. (2023, May 31). How Graphitech Is Using VR4Robots In Nuclear Reactor Decommissioning nucléaires. Retrieved from https://www.graphitech-nuclear.com/en/media/how-graphitech-using-vr4robots-nuclear-reactor-decommissioning-nucleaires

Grebennikova, T., Jones, A., & Sharrad, C. (2021). Electrochemical decontamination of irradiated nuclear graphite from corrosion and fission products using molten salt. *Energy & Environmental Science*, 5501–5512. doi:https://doi.org/10.1039/D1EE00332A

Hanson, D. L., Engel, J. R., & McLaughlin, D. W. (1970). Gamma scanning the primary circuit of the Peach Bottom HTGR. Idaho National Laboratory. Idaho National Laboratory.

M. Hrovat et al. 2008; Fabrication, Properties and Irradiation Performance of Molded Block Fuel Elements; HTR2008-58025

IAEA. (1992). IAEA INSAG-7 (1992). *The Chernobyl Accident: Updating of INSAG-1*. Vienna: International Atomic Energy Agency.

IAEA. (2003). Handling and Processing of Radioactive Graphite from Decommissioning of Nuclear Reactors. *IAEA-TECDOC-409*.

IAEA. (2004). *International Conference on Knowledge Management: Strategies, Information Management and Human Resource Development.* Vienna: Internatioanl Atomic Energy Agency. Retrieved from https://www.osti.gov/etdeweb/servlets/purl/20520786

IAEA. (2006, September). Characterization, Treatment and Conditioning of Radioactive Graphite from Decommissioning of Nuclear Reactors. *IAEA-TECDOC-1521*. Vienna, Austria: International Atomic Energy Agency.

IAEA. (2006). Characterization, Treatment, and Conditioning of Radioactive Graphite from Decommissioning of Nuclear Reactors.. *IAEA-TECDOC-1521*. Vienna, Austria: International Atomic Energy Agency.

IAEA. (2007, January). Use and development of coupled computer codes for the analysis of accidents at nuclear power plants. *IAEA-TECDOC-1539*. Vienna, Austria: International Atomic Energy Agency.

IAEA. (2009). Classification of Radioactive Waste. *IAEA Safety Standards Series No. GSG-1*. Vienna,

Austria: International Atomic Energy Agency.

IAEA. (2011). Disposal of Radioactive Waste. *IAEA Safety Standards Series No. SSR-5*. Vienna, Austria: International Atomic Energy Agency.

IAEA. (2016). Processing of Irradiated Graphite to Meet Acceptance Criteria for Waste Disposal. *IAEA-TECDOC-1790*. Vienna, Austria: International Atomic Energy Agency.

IAEA. (2018). Regulations for the Safe Transport of Radioactive Material. *IAEA SSR-6 (Rev. 1)*. Vienna, Austria: International Atomic Energy Agency.

IAEA. (2021). IAEA Safety Standards. *Series No. SSG-56*. Vienna, Austria: International Atomic Energy Agency.

IAEA. (2024, November). Managing irradiated graphite waste. *IAEA TECDOC - 2072*. Vienna, Austria.

IAEA. (2025). Live Chart of Nuclides. *Nuclear Data Section, Vienna.* Retrieved from https://www-nds.iaea.org/relnsd/vcharthtml/VChartHTML.html

IPPC. (2021). *Climate Change 2021: The Physical Science Basis. Intergovernmental Panel on Climate Change. Contribution of Working Group I to the Sixth Assessment Report.*

JAEA. (2015). Current Status and Issues on Graphite Waste Management in Japan.

KAERI. (2015). Treatment and Disposal Options for Graphite Waste from Korean Research Reactors.

Kawamura et al, 2. (2021). Tritium behavior in isotropic graphite at room temperature. *Fusion Engineering and Design, 172*. doi:https://doi.org/10.1016/j.fusengdes.2021.112801

Kim et al., 2. (2018). *Chemical Decontamination and Packaging of Graphite from MSR Systems.* Journal of Nuclear Fuel Cycle and Waste Technology.

Kingka Tech Industrial Limited. (2024, May 30). Understand The Technology of Heat Pipe Radiators: Principles, Advantages And Applications. Dongguan City, Guangdong Province, China: Kingka Tech Industrial Limited.

Kirby, B. (2011). Brookhaven Graphite Research Reactor (BGRR) D&D Project - 11243. *WM2011 Conference.* Phoenix, AZ.

Kirk, J. S., & Jacobi, J. L. (2016). Disposal of Greater Than Class C Low-Level Radioactive Waste in Andrews County, TX. *WM2016 Conference.* Phoenix, AZ, USA.

Kneitel, T. (2014). BGRR D&D Presentation for the DOE ERAD Working Group. Brookhaven National Laboratory.

Kuhne, L., Lokalisation, F., Freisetzung, & Speziesbestimmung. (2017). *C-14 und H-3 in Reaktorgraphit*. Aachen: Dissertation RWTH.

Laboratoire National Henri Becquerel (LNHB). (2020). Decay Data Evaluation Project (DDEP), Recommended Decay Data for Cesium-137, Version 2020. Nucleide.org. Retrieved from https://www.nucleide.org/DDEP_WG/DDEPdata.htm

Li et al., 2. (2015). *Characterization and Conditioning of Graphite Waste from HTR-10.* Chinese Journal of Nuclear Science and Engineering.

Limer, e. a. (2017, July 10). C-14 Terrestrial Model-Data Comparisons. *Key Issues in Biosphere Aspects of Assessment of the Long-term Impact of Contaminant Releases Associated with Radioactive Waste Management*.

Lindner, E. (2015). *Chemie für Ingenieure, 11*, 258. Wiley-VCH.

Mallants., e. a. (2020). The State of the Science and Technology in Deep Borehole Disposal of Nuclear Waste. *Energies, 13, 833*. doi:https://doi.org/10.3390/en13040833

Marsden, B. (2000). Nuclear graphite for high temeprature reactors. *Gas turbine power conversion systems for modular HTGRs. Report of a technical committee meeting, IAEA-TECDOC-1238*, 177-192. International Atomic Energy Agency (IAEA).

Marsh, H., & Rodríguez-Reinoso, F. (2006). Activated Carbon. *ISBN: 9780080444635.* Elsevier.

McWilliams, A. (2015, September). High Temperature Gas-cooled Reactor (HTGR) Graphite Pebble Fuel: Review of Technologies for Reprocessing. *SRNL-RP-2015-00744*. Aiken, South Carolina, U.S.A.: Savannah River Research Laboratory.

Metcalfe, M., & Tzelepi, A. (2019). The Significance of Carbon 14 in Graphite Reactor Components at End of Generation. *Journal of Environmental Protection, 10*(2), 118-129. doi:https://doi.org/10.4236/jep.2019.102008

Mueller, T., Steele, J., & Gellender, A. (2019). *Environmental Monitoring and Disposal of Radioactive Wastes from U.S. Naval Nuclear-Powered Ships and their Support Facilities.* Department of the Navy.

Murphy, B. D. (2003). ORIGEN-ARP cross-section libraries for Magnox, Advanced Gas-Cooled, and VVER reactor designs. *ORNL/TM-2003/263*. Oak Ridge, TN, U.S.A.: Oak Ridge National Laboratory.

Natesan, K., Purohit, A., & Tam, S. W. (2003, July). Materials behavior in HTGR environments. *NUREG/CR-6824*. U.S. Nuclear Regulatory Commission.

NDA. (2013, September). High Activity Waste - The Long-term Management of Reactor Core Graphite Waste (Gate A). Nuclear Decommissioning Authority.

NDA. (2016, December). Geological Disposal Biosphere Status Report. *Nuclear Decommissioning Authority, NDA Report no. DSSC/454/01*.

NDA. (2020, November 19). Radioactive Waste Strategy September 2019 - Updated 2020. *NDA Radioactive Waste Strategy*. U.K. Nuclear Decommissioning Authority. Retrieved March 26, 2025, from https://www.gov.uk/government/consultations/nda-radioactive-waste-management-strategy/outcome/radioactive-waste-strategy-september-2019

NEA. (2002). Chernobyl: Assessment of Radiological and Health Impacts. Nuclear Energy Agency.

NEA. (2012). International Workshop on the Safe Management and Disposal of Irradiated Graphite. *NEA/RWM/R(2012)2.* Nuclear Energy Agency.

NEA. (2015). Handbook on Lead-bismuth Eutectic Alloy and Lead Properties, Materials Compatibility, Thermalhydraulics and Technologies. Nuclear Energy Agency.

NEA. (2016). Managing Irradiated Graphite Across the Nuclear Life Cycle. *7203*.

NEA. (2022). Advanced Nuclear Fuel Cycles and Radioactive Waste Management. *NEA No. 7471*. Nuclear Energy Agency.

Niephaus D., 'Referenzkonzept zur Direkten Endlagerung von Abgebrannten HTR-Brennelementen in CASTOR THTR/AVR Transport- und Lagerbehältern', Betriebsdirektion Juel-3734, 1999

NRC. (n.d.). 10 CFR 61.55(a)(2)(iv). *Code of Federal Regulations*. US Nuclear Regulatory Commission.

NRC. (1992). NRC Low-Level Waste Performance Assessment Guidance,, 1992. *NUREG/CR-5927*. U.S. Nuclear Regulatory Commission.

NRC. (2000). *A Perfromance Assessment Methodology for Low Level Radioactive Waste Disposal Facilities: Recommendations of NRC'S Performance Assessement Working Group ASSESSMENT METHODOLOGY FOR LOW.* U.S. Nuclear Regulatory Commission. Washington, DC.: U.S. Nuclear Regulatory Commission.

NRC. (2008). *Characterization and Disposal of Irradiated Graphite.* U.S. Nuclear Regulatory Commission.

NRC. (2013). *Assessment of Graphite Structural Integrity under Long-Term Irradiation.* U.S. Nuclear Regulatory Commission.

NRC. (2014). Generic environmental impact statement for continued storage of spent nuclear fuel: Final report. *NUREG–2157, Vol. 2, Appendix B*. Washington, DC, U.S.A.: U.S. Nuclear Regulatory Commission.

NRC. (2019). *10 CFR Part 61: Licensing Requirements for Land Disposal of Radioactive Waste.* U.S. Nuclear Regulatory Commission.

NRC. (2024). § 71.4 Definitions. *10 CFR Part 71*.

NRC. (2024). Safety Evaluation Report. *Docket No. 71-9390, Model No. OPTIMUS® - L Package, Certificate of Compliance No. 9390, Revision No. 2*. Retrieved from https://www.nrc.gov/docs/ML2334/ML23345A096.pdf

OECD/NEA. (2014). *Radioactive Waste Management Programmes in OECD/NEA Member Countries - France.* OECD/NEA. Retrieved from https://www.oecd-nea.org/rwm/profiles/France_profile_web.pdf?utm_source=chatgpt.com

OECD/NEA. (2016). Thermochemical Treatment of Graphite Waste. *Nuclear Science NEA/CSNI/R(2016)1*.

Pang, e. a. (2021, February 23). Liquid-phase oxidative digestion method for radioactively contaminated carbon-containing material. *United States Patent 10,930,406*. United States of America.

Pappano, P., Strizak, J., Burchell, T., & Albers, T. (2009). Deep Burn: A Hot New Way to Reuse Irradiated Graphite. *10th International Nuclear Graphite Specialists Meeting.* Yellowstone, U.S.A.

Park et al., 2. (2017). *Waste Management Strategy for VHTR Graphite.* KAERI Technical Report.

Penney, e. (2017). Report on the accident at Windscale No. 1 Pile on 10 October 1957. *Journal of Radiological Protection, 37*, 780–796.

Perera, J. (2024, April 11). The complexities of irradiated graphite. *Nucelar Engineering International*.

Pierson, H. (1993). *Handbook of Carbon, Graphite, Diamonds and Fullerenes*. Noyes Publications.

Pina G., Decontamination Factors by Decontamination Agents under Normal Pressure, CARBOWASTE Deliverable D-4.3.5, 2008

Pina, G. (2021, October 3). Graphite Management to Enhance Disposability by Achieving Compliance with LILW Disposal WAC for El Gabril. *Eurad*.

Plätzer S. and Mielisch M., Unloading of the Reactor Core and Spent Fuel Management of THTR, IAEA TECDOC-1043, 1998

Poncet, B., & Petit, L. (2013). Method to assess the radionuclide inventory of irradiated graphite waste from gas cooled reactors. *J. Radioanal. Nucl. Chem, 298*, 941-953.

Powers, D. (2001, October 10-12). Trip Report—Travel by D. A. Powers to Attend the High-Temperature Gas-Cooled Reactor Safety and Research Issues Workshop. Rockville, MD, U.S.A.: US NRC, ADAMS Accession Number: ML020450645.

Qualls, A. (2018, September). Advanced Reactor Technology Program Molten Salt Reactor Campaign FY 2018 Summary. *ORNL/SPR-2018/987*. Oak Ridge, TN, U.S.A.: Oak Ridge National Laboratory.

Rasson, e. (2022, January 18). Generic method to assess transmutation feasibility for nuclear waste treatment and application to irradiated graphite. Retrieved March 14, 2025, from https://arxiv.org/pdf/2201.02623

Reynolds, S., 2025; ' UKAEA's Experience Detritiating and Treating Radioactive Graphite and CFC Fusion Wastes' *presented at* Nuclear Industry Association Symposium, The Engine Rooms, Risley, UK, April 30th 2025

Rizzato C., von Lensa W., Vulpius D., Baginski K., and Fischer C., Thermal Treatments of Irradiated Graphite, Joint Workshop IAEA/CARBOWASTE – Lithuania, 2013

Sato, H. E. (2013). *Irradiated Graphite Waste Management at the HTTR.* Tokyo: JAEA.

Schlögl, B., 'Oxidationskinetik Innovativer Kohlenstoffmaterialien Hinsichtlich Schwerer Lufteinbruchstörfälle in HTR's und Graphitentsorgung oder Aufarbeitung', Schriften des Forschungszentrums Jülich, Reihe Energie & Umwelt / Energy & Environment, ISSN 1866-1793, Band / Volume 89, 2010)

Schmutz, H., Sabharwall, P., & Stoots, C. (2012). *Tritium Formation and Mitigation in High Temperature Reactors.* Idaho National Laboratory. Idaho Falls: Idaho National Laboratory.

Schneider-Eickhoff R. and Stuke M., Dry Storage of THTR Spent Fuel in Germany, BGZ, 2024 Workshop on Storage and Transportation of TRISO and Metal Spent Nuclear Fuels, December 3rd to 5th, 2024

Schweitzer, D., Gurinsky, D., Kaplan, E., & Sastre, C. (1987, September). A Safety Assessment of the Use of Graphite in Nuclear Reactors Licensed by the U.S. NRC. *NUREG/CR-4981*.

SKB. (2021, February). Our method of final disposal. Solna, Sweden. Retrieved March 19, 2025, from https://skb.com/future-projects/the-spent-fuel-repository/our-methodology/

Srinivasan, M. (2011, April 5). Advanced Reactor Research Plan and Status for Graphite Materials. *The Minutes of the ACRS Future Plant Designs Subcommittee*. US NRC ML111110676.

State of Utah, Department of Environmental Quality. (2021, September 9). Waste Management and Radiation Control Board Meeting. Salt Lake City, Utah, U.S.A. Retrieved 6 3, 2025, from https://www.utah.gov/pmn/files/775181.pdf

Sterner, H., & Rittscher, D. (2009, May 29). The AVR (HTGR) Decommissioning Project With New Strategy. 545-550. doi:https://doi.org/10.1115/ICEM2007-7040

Stevens, D. J., & Marsh, H. (1987). Chemistry of Graphite Oxidation. *Carbon(6), 25*, 747-756.

Stithem, A. (1993). *Tritium leach test on H-327 graphite*. Retrieved March 20, 2025, from nrc.gov: https://www.nrc.gov/docs/ML2011/ML20114D687.pdf

Supko, E. M., & Schwartz, M. H. (2011). *Overview of High-Level Nuclear Waste Materials Transportation: Processes, Regulations, Experience and Outlook in the U.S.* Washington, D.C. U.S.A.: Energy Resources International, Inc.

Takada, H. e. (2012). *Tritium Behavior in Irradiated Graphite.* Journal of Nuclear Science and Technology.

Takeshita, K., Nakano, Y., Shimizu, M., & Y, F. (2002). Recovery of 14C from Graphite Moderator of Gas-Cooled Reactor. *Journal of Nuclear Science and Technology, 39*(11), 1207-1212.

Tollefson, J. 5.–1. (2014). US seeks waste-research revival. *Nature*, 15-16. doi:https://doi.org/10.1038/507015a

University of Manchester. (2025). Novel graphite treatment 'could reduce UK nuclear waste burden'. Retrieved March 19, 2025, from https://www.manchester.ac.uk/about/news/novel-treatment-technology-could-reduce-uk-nuclear-waste-burden/

Urben, P.G. (Ed). (2006). Bretherick's Handbook of Reactive Chemical Hazards. *7th Ed.*

VATESI. (2024). *Lithuanian National Report Under the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management.* State Nuclear Power Safety Inspectorate (VATESI). Retrieved from https://vatesi.lrv.lt/public/canonical/1723461994/2113/7th_National_Report_Lithuania2024.pdf

Venugopal, S., & Prasad, K. G. (1983). Decomposition of Graphite with Nitric Acid. *Journal of Nuclear Materials, 119(2-3)*, 153–161

Verfondern K., R&D Works on HTR Spent Fuel Treatment in Germany, IAEA TM on the Management of Spent Fuel (Pebbles and Compacts) from High Temperature Reactors, July 7-11, 2025, article in press

Verfondern K., Pohl, P. *et al*., 'Characterization of Spent HTGR Fuel in Germany, European RAPHAEL Project', D-BF1.2, 2006 https://snetp.eu/repository/

von Lensa, W., Cardinal, G., Bradbury, D., Eccles, H., Fachinger, J., Grambow, B., G., P. (2008). Treatment and Disposal of Irradiated Graphite and other Carbonaceous Waste. *Proceedings of the 4th International Topical Meeting on High-Temperature Reactor Technology.* Washington D.C.

Vulpius D., Baginski K., Fischer C. and Thomauske B., Location and chemical bond of radionuclides in neutron-irradiated nuclear graphite, Journal of Nuclear Materials 438 (2013), page 163–177, Figure 8

Wahlen, E., Wahl, J., & Pohl, P. (2000). Status of the AVR decommissioning project with special regard to the inspection of the core cavity for residual fuel. *WM'00 Conference.* Tucson, AZ.

Wellenreuther, H., & Wirtz, R. (2011). Disposal of Irradiated Graphite from AVR. *International Conference on Management of Spent Fuel from Nuclear Power Reactors, IAEA*. Vienna, Austria: International Atomic Energy Agency

Wenzel U. *et al*., Determination of 14 C in Spent HTGR Fuel Elements, Journal of Radioanalytical Chemistry, Vol. 53, No. 1-2, 1979

Wickham, A. (1989). *Caring for the Graphite Cores.* London: Inst. of Mech. Engineers.

Wickham, A. J., Von Lensa, W., Srinivasan, M., & Desrosiers, A. (2025). TLR 1 - Survey of Graphite Waste.

Wickham, A., & Bradbury, D. (2006, March). Graphite Decommissioning: Options for Graphite Treatment, Recycling, or Disposal, including a discussion of Safety-Related Issues. *EPRI Report: 1013091*. Palo Alto, CA, U.S.A.: Electric Power Research Institute.

WICKHAM A.J. and BRADBURY D., 2007, 'Graphite Dust Deflagration', EPRI Palo Alto CA, 1014797 and 1015460

WICKHAM A.J. and BRADBURY D., 2010; 'C-14 in Irradiated Graphite Waste', EPRI 1021109

Wickham, A., & Rahmani, L. (2010, October). Graphite Dust Explosibility in Decommissioning: A Demonstration of Minimal Risk. *IAEA TECDOC-1647*. Vienna, Austria: International Atomic Energy Agency.

Wickham, A., Meyer, W., & O'Sullivan, P. (2019). Irradiated Graphite Processing Approach: Update on Project 'GRAPA'. *The 20th International Nuclear Graphite Specialist Meeting.* Bruges, Belgium.

Wickham, A., Steinmetz, H.-J., O'Sullivan, P., & Ojovan, M. I. (2017). Updating irradiated graphite disposal: Project 'GRAPA' and the international decommissioning network. *Journal of Environmental Radioactivity, 171*, 34-40. doi:https://doi.org/10.1016/j.jenvrad.2017.01.022

Worth R.N., Theodosiou A., Bodel, W., Arregui-Mena J.D., Wickham A.J., Jones A.N. and Mummery, P.M., 2021; 'The Distribution and Selective Decontamination of Carbon-14 from Nuclear Graphite', J. Nuclear Materials, DOI: https://doi.org/10.1016/j.jnucmat.2021.153167

Yang, E. A. (2004). Destruction characteristics of irradiated graphite waste at various waste thermal conditions. *Waste Management and the Environment II*. WIT Press, www.witpress.com. Retrieved March 26, 2025, from https://www.witpress.com/Secure/elibrary/papers/WM04/WM04010FU.pdf

Zhang et al., 2. (2014). *Experience from HTR-10 Operation and Its Implication for HTR-PM.* Nuclear Engineering and Design.

Zhang, E. (2021). Sodium-Potassium Alloy Heat Pipe under Geyser Boiling Experimental Study: Heat Transfer Analysis. *Energies, 14, 7582*. doi:https://doi.org/10.3390/en14227582

Zhou, T. (2022, May). Review of Potential Wigner Effect Impacts on the Irradiated Graphite in Decommissioned Hanford Reactors. Richland, WA. U.S.A.: Central Plateau Clean Up Company.

APPENDIX A. THE PURITY OF NUCLEAR GRADE GRAPHITE AND NEUTRON ACTIVATION PRODUCTS RESULTING FROM REACTOR OPERATION

The very high temperature used for the manufacture of graphite, around 3000 °C, used in nuclear reactor applications provides material of very high purity. At ~ 3000 °C, many impurities present in carbon precursors (like metals, sulfur, nitrogen) either vaporize (due to low volatility at high temperatures) or migrate to surfaces. While graphite itself does not "melt" at this temperature (it sublimes at about 3300K at atmospheric pressure), localized surface diffusion and recrystallization (analogous to zone refining effects) can occur. High temperature promotes segregation of impurities, pushing them toward less stable regions (surfaces), which can then be physically removed or vaporized. Graphitization involves a multi-mechanism purification process, where zone refining-like segregation, high-temperature sublimation, and surface cleaning all work together to improve purity.

Purification Mechanisms

Volatilization: Largely, the volatilization of the impurities is the mechanism for purification. For example, sulfur vaporizes as SO_2 or elemental S. Nitrogen is released as N_2, HCN, etc. Hydrogen is eliminated as H_2 gas. Some metals (e.g., zinc, cadmium, lead, alkali metals) can either vaporize directly or form volatile carbides/oxides. This natural "distillation" effect greatly reduces impurity content.

Zone refining: Graphitization involves a rearrangement of disordered carbon (turbostratic carbon) into ordered graphitic layers. As graphitization progresses, impurities (especially metals, silicon, or non-volatile elements) tend to be excluded from the perfect crystal lattice. These rejected impurities migrate to graphite grain boundaries or interstitial defects — like how impurities are rejected into a molten zone boundary during classic zone refining. At these boundaries, they can be more easily removed by gas-phase reactions (like high-temperature halogen treatment), mechanical processing (like shaving of the outer layer of the graphitized product), or surface cleaning.

Carbothermal reduction of oxides: If metal oxides are present (e.g., Fe_2O_3, SiO_2, Al_2O_3), these can react with carbon at high temperatures, forming volatile species like, (a) SiO (gas) from SiO_2 and CO (gas) from carbon reacting with oxygen contamination. This helps to remove oxygen-containing impurities.

Internal Diffusion and Recrystallization: At 3000 °C, carbon atoms become very mobile, and the graphite lattice rearranges into nearly perfect ABAB stacking (ideal graphite). During this process, impurities trapped in defects or turbostratic regions get pushed out — either vaporizing or migrating to boundaries where they concentrate (like how molten zones sweep impurities in classic zone refining).

Graphite Surface Purification

The surface of the graphite can also undergo purification by vaporizing loosely bound contaminants. Gas-phase carbon species, like C3 or C2 molecules, can form, which can carry away contaminants. Surface etching by controlled atmospheres (vacuum, argon, or reactive gases like chlorine) can also remove surface-bound metals.

Graphite manufacturers have traditionally used chlorine gas in order to remove boron and heavy metals as their volatile chlorides. This was replaced with freons (also halogen compounds), but its use ceased with the realization of its effects on the ozone layer. More recently, magnesium fluoride has been employed (useful because it contains nothing which can subsequently be activated under irradiation): some manufacturers of future nuclear grades are nonetheless still employing chlorine for purification.

A summary of the above phenomena involved in graphitization purification is provided in Table 16.

Table 16. A Summary of the Purification Mechanisms During Graphitization

Mechanism	What happens at 3000 °C?	Impurities removed
Sublimation	Direct vaporization of volatiles	S, N, H, Zn, Pb, Cd
Segregation to boundaries	Impurities expelled from crystal lattice	Metals, Si
Carbothermal reduction	Metal oxides reduced to metals or gases	SiO_2, Al_2O_3
Diffusion & recrystallization	Rearrangement to near-perfect graphite	Internal defects purged
Surface purification	Surface volatilization and etching	Surface-bound metals

Purity requirements for nuclear graphite: (Marsden, 2000) has identified several impurities as incompatible for reactor operation, and final decommissioning and disposal. In an earlier ASTM nuclear graphite material specification (ASTM, 2005), these impurities were classified into five categories. This information is summarized below:

1. Neutron absorbing impurities (NAM): boron (B), cadmium (Cd), dysprosium (Dy), Europium (Eu), Gadolinium (Gd), Mercury (Hg), Samarium (Sm), and tungsten (W).
2. Oxidation promoting catalysts (OPC): aluminum (Al), barium (Ba), calcium (Ca), copper (Cu), iron (Fe), lead (Pb), magnesium (Mg), manganese (Mn), nickel (Ni), potassium (K), silicon (Si), silver (Ag), sodium (Na), sulfur (S), titanium (Ti), and vanadium (V).
3. Activation relevant impurities (ARI): cadmium (Cd), calcium (Ca), chlorine (Cl), cobalt (Co), europium (Eu), mercury (Hg), and samarium (Sm).
4. Metallic corrosion relevant impurities (MCRI): chlorine (Cl) and sulfur (S).
5. Fissile/Fissionable element (FFE): titanium (Ti) and uranium (U).

Notably, the ash content limits were also prescribed, for “low-purity” as 1000 ppm and for “high-purity” graphite as 300 ppm. The high-purity graphite should have boron equivalent content of less than 2 ppm, and the low-purity graphite should have boron equivalent content greater than 2 ppm but less than 10 ppm.

Analysis of “Historical” and “Modern” Nuclear Grade Graphites

It is pertinent to review and evaluate the purity of the nuclear graphites used in previous reactors to establish information on activation products during service and determine their potential effects in graphite waste management. A significant characteristic of graphite for decommissioning and disposal is the initial “ash content” of a manufactured graphite. Ash content refers to the residual inorganic material left behind after the graphite is completely combusted in air or oxygen at high temperatures (typically around 800-1000 °C). This leftover material — the “ash” — consists of impurities that were present in the raw materials or introduced during manufacturing. For nuclear graphites historically used in reactor cores (especially those developed pre-1980), the typical design requirement was to keep ash content below 500 ppm. This was not because every graphite hit the same number — rather, it was a general target threshold set by reactor designers to limit neutron absorption and avoid unwanted impurities (like boron, which is a neutron poison). The ash can contain various metal oxides and other inorganic compounds, as shown in Table 17.

Table 17. Major Elements in Ash from Graphite (Inorganic Compounds).

Element	Common Source	Impact
Silicon (Si)	Petroleum coke raw material	Affects mechanical properties
Iron (Fe)	Contamination from processing equipment	Can catalyze unwanted reactions
Calcium (Ca)	Processing additives	Influences thermal properties
Boron (B)	Natural impurity in raw carbon	Strong neutron absorber
Sodium (Na)	Environmental contamination	Causes chemical instability
Phosphorus (P)	Coke impurity	Weakens graphite structure
Sulfur (S)	Petroleum coke impurity	Influences thermal conductivity

Besides the post-reactor life issues for graphite management, low initial ash content is important as a structural material (support components) and as a neutron moderator.

A summary of typical ash contents of manufactured graphite is shown in Table 18.

Table 18. Typical Ash Content Ranges for Graphites

Application	Target Ash Content
Commercial non-nuclear graphite	500 - 3000 ppm
Nuclear graphite (historic grades)	< 500 ppm
High-purity nuclear graphite (modern HTGR/MSR/fusion)	< 200 ppm, sometimes < 100 ppm

The measured ash content can vary by batch. The actual ash content could be significantly lower — some high-purity graphites (like IG-110, PCEA, and modern Japanese grades) often test below 200 ppm, sometimes even near 50 ppm. For older graphites (like Magnox-era grades), ash content tended to hover closer to the upper limit (~ 400-500 ppm). But detailed, published ash content data for each specific product or batch is not always available — so "< 500 ppm" may be used as a sort of default, conservative value. Modern graphites (post-1980) intended for HTGR and fusion reactors often pushed for ash contents below 200ppm (sometimes below 100 ppm for advanced designs). These graphites undergo high-temperature halogen (or halogen compound) treatments after graphitization, sometimes twice to achieve such high purity.

A comparison of purity of nuclear graphites used as GCC in HTGRs is provided in Table 19. The most significant property, in terms of post-reactor operation, decommissioning, and disposal of irradiated graphite is the total ash content, which is representative of graphite's purity. As seen in Figure 5, the most recent grades have several times less ash content than experimental and commercial reactors of the past.

Table 19. A Comparison of Impurities in Nuclear Grade Graphites[20]

Graphite Grade	Carbon Content (%)	Ash Content (%)	Major Impurities	Density (g/cm³)	Applications
H-451	99.5	0.3	Si, Fe, Al, Ca	1.76	GCC
H-327	99.5	0.48 (max observed, but specification is < 0.1%)	B, Fe, V, Ti	1.78	GCC
ATR-2E	99.8	0.15	B, Si, Fe, Al	1.83	GCC
Gilsocarbon	99.7	0.2	Si, Fe, Ti, Ca	1.85	Nuclear reactor bricks (AGR)
NGB-18	99.9	0.05	B, Si, Al	1.87	GCC
IG-110	99.9	0.05	B, Si, Fe	1.77	GCC
PCEA	99.8	0.1	B, Si, Fe, Ca	1.84	GCC
Mersen nuclear graphite	99.9	0.05	B, Si, Fe	1.86	Nuclear Applications
ET-10	≥ 99.99	≈ 0.01	Si, Fe, Al, Ca, Mg (low ppm)	1.77–1.80	Nuclear research, moderators, neutron targets
ETU-10	≥ 99.9995%	≈ 0.0005%	Nearly free of detectable metals	1.80–1.83	High-purity nuclear R&D, isotope production targets, irradiation capsules

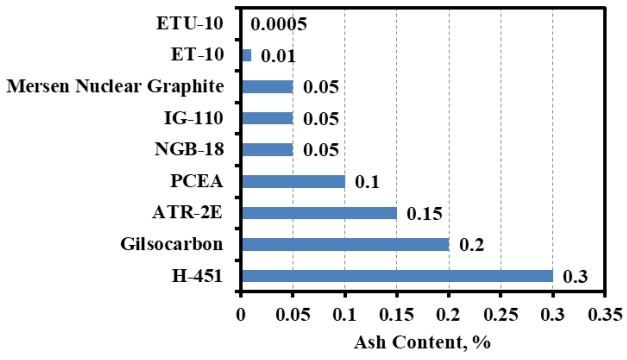

Figure 5. The Ash Content of Various Nuclear Graphite Grades

[20] Property values were excerpted from a variety of sources.

The following general references were also consulted for writing this Appendix.

Inagaki, M. (2000). New Carbons: Control of Structure and Functions. Elsevier.

IUPAC Recommendations (1995). Definitions Relating to Characterization and Properties of Carbon as a Solid. Pure and Applied Chemistry, 67(3), 473-506.

Jenkins, G.M., & Kawamura, K. (1976). Polymeric Carbons – Carbon Fibre, Glass and Char. Cambridge University Press.

Kinoshita, K. (1988). Carbon: Electrochemical and Physicochemical Properties. Wiley.

Marsh, H., & Rodríguez-Reinoso, F. (2006). Activated Carbon. Elsevier.

Mochida, I., et al. (1990). High Temperature Graphitization and its Effect on Carbon Structure. Carbon, 28(3), 311-320.

Pierson, H.O. (1993). Handbook of Carbon, Graphite, Diamonds and Fullerenes. Noyes Publications.

Schlogl, R., & Ertl, G. (1990). Carbon as a Catalyst Support: Graphitization and Surface Reactivity. Carbon, 28(6), 841-851.

Singer, L.S., & Huttinger, K.J. (1990). The Influence of Heat Treatment on the Properties of Synthetic Graphite. Carbon, 28(3), 321-330.

Toyota, H., & Oya, A. (1984). High-Temperature Purification of Carbon Materials. Carbon, 22(5), 377-381.

APPENDIX B. DECAY CHARACTERISTICS OF SOME COMMONLY FOUND RADIOISOTOPES IN IRRADIATED GRAPHITE

This Appendix provides a detailed analysis of isotopes found in irradiated nuclear graphite, including their origins, reactor details, irradiation conditions, decay models, and mathematical formulations.

Isotope Origins and Reactor Details

Carbon-14 (^{14}C): It is formed in nuclear reactors within the graphite moderator primarily through the neutron activation of nitrogen impurities (from quinolines in the original pitch) and from the 1.1% naturally occurring ^{13}C. ^{14}C is generated via neutron activation reactions: $^{14}N(n, p)^{14}C$ and $^{13}C(n, \gamma)^{14}C$ plus, to a lesser extent, reactions with oxygen $^{17}O(n,\alpha)^{14}C$, this last especially where coolants containing carbon dioxide or air have been used (Wickham and Bradbury 2010).

Tritium (^{3}H): It is produced via neutron activation of lithium impurities or from ternary fission reactions within the reactor. Tritium is mainly generated from ^{6}Li impurities in the graphite and A3-matrix by the $^{6}Li(n,\alpha)^{3}H$ reaction from and, in HTGR, via the activation of the ^{3}H isotope in the helium coolant by the $^{3}He(n, p)^{3}H$ reaction. ^{3}H is present in both graphite-moderated reactors and heavy-water reactors. The ash content of the reactor graphite is maintained at minimal levels to assist the reduction of tritium production. The accumulated irradiation dose from tritium varies based on reactor type and operational history.

Cobalt-60 (^{60}Co): ^{60}Co results from neutron activation of cobalt impurities present in reactor materials or introduced via wear and corrosion products, along with a contribution from other transition metals. It is observed in various reactor types, including pressurized water reactors (PWRs) and boiling water reactors (BWRs). It is not directly related to graphite but can be present if cobalt impurities exist. Regarding nonirradiated ash content, the graphite specification requires limits in cobalt content to reduce ^{60}Co formation. The accumulated irradiation dose from ^{60}Co depends upon the presence of relevant precursors and the neutron flux.

Chlorine-36 (^{36}Cl): It is formed through neutron activation of chlorine impurities within the reactor environment and also arises from chlorine picked up from the air as the compound HOCl. It can be present in various reactor types. The nonirradiated ash content should have low chlorine content to minimize ^{36}Cl production. The accumulated irradiation dose varies based on chlorine impurity levels and neutron exposure. The maximum irradiation temperature depends on reactor design and operational conditions: inorganic forms of chlorine compound tend to be lost during reactor operation, leaving organic forms associated with the graphite waste form.

Strontium-90 (^{90}Sr) and Cesium-137 (^{137}Cs): Both are fission products resulting from the splitting of uranium or plutonium isotopes during reactor operations. These are present in all reactors where fission occurs. These are not directly related to graphite composition but can adsorb onto graphite surfaces. The nonirradiated ash content is not applicable as they are fission products. The relationship to accumulated irradiation dose is proportional to reactor burn-up and fission events.

Decay Models and Equations

The decay of radioactive isotopes is governed by first-order kinetics, described by the equation:

$$N(t) = N_0 \times e^{(-\lambda t)},$$

where:

N(t) is the quantity of the isotope at time t, N_0 is the initial quantity at t=0, λ is the decay constant, related to the half-life, $(t_{1/2})$ by $\lambda = \frac{\ln(2)}{t_{1/2}}$.

Estimated decay characteristics of some common radioisotopes in irradiated graphite are shown in Figure 6 and have been derived from published decay data.

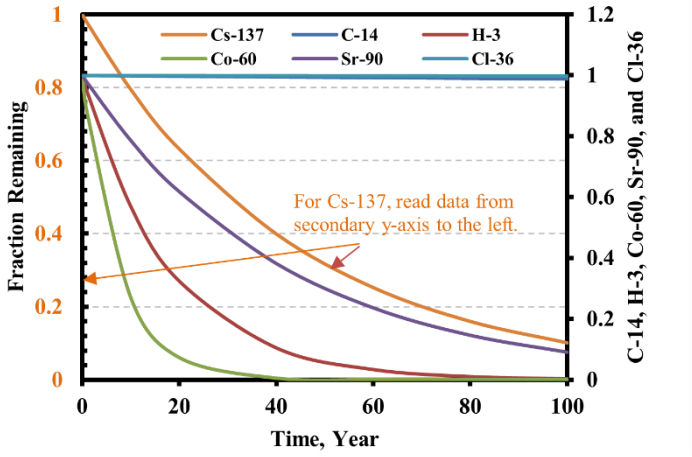

Figure 6. A Depiction of Decay Characteristics of Common Radioisotopes in Irradiated Graphite

Annotated Data Sources:

- ^{14}C : IAEA, Technical Reports Series No. 409 (IAEA, 2003)
- ^{3}H : IAEA, Nuclear Data Sheets (IAEA, 2025)
- ^{60}Co : Decay Data Evaluation Project (DDEP) (Laboratoire National Henri Becquerel (LNHB), 2020)
- ^{36}Cl : IAEA Live Chart of Nuclides (IAEA, 2025)
- ^{90}Sr: Nuclear Data Sheets, Vol. 129 (2015)
- ^{137}Cs: IAEA, Technical Reports Series No. 472 (IAEA, 2011)

Estimating the Total Activity of Carbon-14 (^{14}C) in Irradiated Graphite

The estimation steps involve determining the specific activity of ^{14}C within the graphite and then scaling this value by the mass of the graphite sample. Here is a step-by-step guide to this estimation:

1.1 Measuring Specific Activity

The specific activity of ^{14}C (activity per unit mass) in graphite can be determined through radiochemical analysis. A rapid analysis method involves oxidizing small graphite samples (ranging from 1 to 100 micrograms) to extract ^{14}C, which is then measured to determine its specific activity (Garankin, et al., 2018).

1.2 Calculating Total Activity

Once the specific activity (A_s) is known, the total activity (A_t) of a given mass (m) of irradiated graphite can be calculated using the formula:

$$A_t = A_s x\ m,$$

where:

- A_t = Total activity of ^{14}C in the sample (in becquerels, Bq)
- A_s = Specific activity of ^{14}C (in becquerels per kilogram, Bq/kg)
- m = Mass of the graphite sample (in kilograms, kg)

1.3 Example Calculation

Assume the specific activity of ^{14}C in a particular irradiated graphite sample is determined to be 5,000 Bq/kg, and the mass of the graphite is 100 kg. The total activity would be:

A_t = 5,000 Bq/kg x 100 kg = 500,000 Bq.

1.4 Factors Influencing ^{14}C Activity

Several factors can affect the specific activity of ^{14}C in irradiated graphite:

- Neutron Flux and Energy Spectrum: Higher neutron fluxes and specific energy spectra can increase ^{14}C production.
- Irradiation Time: Longer exposure periods lead to higher ^{14}C accumulation.
- Graphite Purity: Impurities, especially nitrogen, can contribute to additional ^{14}C production through neutron interactions alongside that produced from naturally present ^{13}C.

1.5 Importance of Accurate Measurement

Accurate determination of ^{14}C activity – and its chemical form - is crucial for the safe disposal and management of irradiated graphite waste. ^{14}C is a long-lived radionuclide, and its activity significantly influences the classification and handling of radioactive waste (Metcalfe & Tzelepi, 2019).

In summary, estimating the total of ^{14}C activity in irradiated graphite requires precise measurement of its specific activity through radiochemical methods, followed by scaling this value according to the mass of the graphite. Understanding the factors that influence ^{14}C production can aid in making more accurate estimations and inform effective waste management strategies.

APPENDIX C. WIGNER ENERGY, GRAPHITE OXIDATION MECHANISMS, AND THE BEHAVIOR OF GRAPHITE IN DUST FORM

Recent preliminary work on dismantling graphite-moderated reactors, particularly in the U.K. and in Lithuania, has shown that concerns are regularly raised in the context of the propensity of graphite to accumulate (and potentially release) so-called Wigner (internal) energy arising from neutron bombardment during operation, and also the potential of graphite to participate in fire situations and also of the graphite dust to produce a deflagration (dust explosion). All of these issues have been comprehensively investigated, and in the context of future reactor systems can readily be shown to be either inconsequential or easily mitigated during graphite handling and reactor dismantling. These matters were comprehensively discussed in a previous report (Wickham A.J. et al. 2025) but are principally of concern at the dismantling and handling stage, and so additional information is provided here.

Neutron bombardment causes damage to the graphite's crystal structure, and the accumulation of stored energy from displaced atoms in reactors in which the graphite was at relatively low temperatures (less than 200 °C) and also causes volumetric changes that increase the fragility at high radiation doses (greater than 1 to 3 displacements per atom (dpa)). Fortunately, Wigner energy concerns are largely limited to older reactors (for example, the U.K. Windscale Piles, BGRR) that operated at low temperatures; the phenomenon is well understood and in those cases could be mitigated by controlled heating to avoid any spontaneous energy release (Gallego, Burchell, & Srinivasan, 2009), (Schmutz, Sabharwall, & Stoots, 2012) (Wickham A. , 1989). The release characteristics of Wigner energy as a function of increasing temperature are well understood and are documented in the references cited. The graphite operating temperatures in HTGRs and MSR will be such that no significant Wigner energy accumulation will occur in the graphite.

Graphite is chemically stable and only slowly oxidizes (chemically) under normal conditions, which helps in handling and storage. It cannot burn in air at temperatures below 3300K because there is no significant vapor pressure under ambient conditions. Descriptions of "graphite fires" – e.g. at Chernobyl – are misplaced, as discussed in more detail below.

Graphite is actually used as a fire *extinguishant* because of its high heat capacity[21], and is in regular use in powdered or exfoliated form to extinguish metal fires, particularly in nuclear post-irradiation examination facilities. In regard to the potential for a graphite dust deflagration, the standard ISO test for graphite dust identifies it as 'mildly explosive' but this test is conducted in the presence of a high-energy ignition source. However, for graphite dust generated during reactor dismantling, industry experience shows this risk can be managed with proper precautions and is generally low (Andrews & et.al, 2005), (Wickham and Bradbury 2007). Six conditions have to be simultaneously satisfied in order to initiate a deflagration: a combustible substance (as graphite clearly is), an oxidizing atmosphere (i.e. air), and an ignition source (flame or electrical spark, for example): in addition there has to be turbulent flow (to keep the dust in suspension, and both the particle size and concentration have to be with certain ranges to permit the propagation of a flame front). In any reasonable dismantling and waste handling scenario, at least one, and maybe more, of these conditions can be avoided and hence the risk is extremely small. In earlier U.S. work in support of this contention, combustion and deflagration feasibility were analyzed considering the minimum ignition energy

[21] An excellent demonstration has made of heating a sample of graphite with an oxy-acetylene torch flame to white heat. Upon cutting off the acetylene and leaving a pure oxygen jet, it has the effect of cooling the graphite down, accompanied by a decreasing rate of chemical oxidation to carbon dioxide as the temperature falls.

required, and the combination of fuel, oxidant, and ignition source (Srinivasan, 2011). Combustion always occurs in the vapor phase; liquids are volatilized, and solids need to be vaporized prior to combustion. This interaction is depicted in the Occupational Safety and Health Administration (OSHA) combustion triangle (Figure 7).

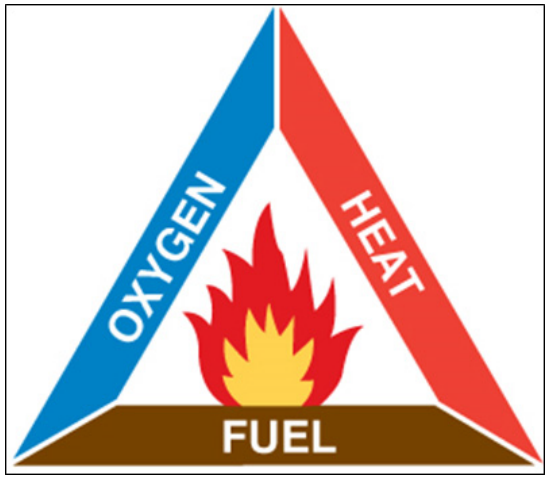

Figure 7. Requirements for Combustion

The propensity for reaction can also be ascertained from consideration of minimum ignition energy (MIE). Table 20 provides MIE values for some common materials in vapor/gas form and in the dust cloud form (Ebadat, 2009) and for nuclear graphite (Wickham & Rahmani, 2010).

Table 20. Relative Comparison of Ignition Energies for Explosibility

Atmosphere	**Material**	**Minimum Ignition Energy (MIE), Joules**
Vapor/gas	Propanol	0.00065
	Ethylene acetate	0.00046
	Methane	0.00028
	Propane	0.00025
	Ethane	0.00024
	Methanol	0.00014
	Acetylene	0.000017
	Hydrogen	0.000016
	Carbon disulfide	0.000009
Dust cloud	PVC	2
	Zinc	0.2
	Wheat flour	0.05
	Polyethylene	0.03
	Sugar	0.03
	Magnesium	0.02
	Sulfur	0.015
	Aluminum	0.01
	Epoxy resin	0.009
	Zirconium	0.005
	Nuclear graphite	**10,000-20,000**

Vapors and gases have extremely low MIEs, indicating high flammability. For instance, carbon disulfide requires only 0.000009 Joules to ignite. Dust clouds, while typically less easily ignited, still pose hazards at low MIEs—particularly materials like aluminum (0.01 Joules)[22] and magnesium (0.02 Joules). Organic dust, such as wheat flour and sugar, has MIEs around 0.03 to 0.05 Joules, underscoring the risks posed by food processing and agricultural facilities. Metal dust (magnesium, aluminum, zirconium) exhibits particularly low ignition energies, reinforcing their well-known explosibility risks in fine particle form.

Nuclear graphite stands out, requiring vastly higher energy (10,000-20,000 Joules), which highlights its low combustibility under normal conditions.

The propensity for flammability of nuclear graphite can also be compared with that of different coals as a carbon family as shown in Table 21. The data for coals were obtained from (Lindner, 2015).

[22] It is noted that the Windscale reactor had an aluminum container.

Table 21. Chemical Composition of Coals and Nuclear Graphite

Coal Type	Volatiles (%)	Carbon (%)	Hydrogen (%)	Oxygen (%)	Sulfur (%)	Heat Content (kJ/kg)
Braunkohle (lignite)	45-65	60-75	6.0-5.8	34-17	0.5-3	<28470
Flammkohle (flame coal)	40-45	75-82	6.0-5.8	>9.8	~1	<32870
Gasflammkohle (gas flame coal)	35-40	82-85	5.8-5.6	9.8-7.3	~1	<33910
Gaskohle (gas coal)	28-35	85-87.5	5.6-5.0	7.3-4.5	~1	<34960
Fettkohle (fat coal)	19-28	87.5-89.5	5.0-4.5	4.5-3.2	~1	<35380
Esskohle (forge coal)	14-19	89.5-90.5	4.5-4.0	3.2-2.8	~1	<35380
Magerkohle (non-baking coal)	10-14	90.5-91.5	4.0-3.75	2.8-3.5	~1	35380
Anthrazit (anthracite)	7-12	>91.5	<3.75	<2.5	~1	<35300
Nuclear graphite irradiated at 30 °C	0	>99.99	0	0	0	<3,000
Nuclear graphite irradiated at 450 °C	0	>99.99	0	0	0	<75

The classification of coal is generally based on the content of volatiles. However, the exact classification varies between countries. Graphite, after greater than 2800 ° C graphitization, has just carbon as its chemical constituent. That is, there are virtually no volatiles, hydrogen, or oxygen (fuel source). The stored energy of 30 °C irradiated nuclear graphite is less than 3,000 kJ/kg. This energy can be released upon heating, potentially leading to energy releases of up to 3,000 kJ/kg. However, at an irradiation temperature of 450 °C, the graphite's structure allows for the annealing of defects during irradiation, resulting in much lower accumulation of stored energy of less than 75 kJ/kg (Gallego, Burchell, & Srinivasan, 2009). This heat content can be negligible for combustion. Therefore, sustained ‘burning’ in the presence of limited oxygen and ignition source is not favored. Comparatively, the heat content of nuclear graphite can be negligible, by a focus on the structural integrity.

Despite the above technical aspects, there has been a perennial concern about the alleged “burning” of graphite due to Windscale and Chernobyl graphite reactor accidents. In this respect, and for better public outreach, it is worthwhile to summarize the differences between these reactor types and the modern HTGRs. The salient features of the Windscale and Chernobyl reactors are provided in Table 22. Selected features of Windscale and Chernobyl graphite-moderated reactors.

Table 22. Selected features of Windscale and Chernobyl Graphite Moderated Reactors

Type	Windscale	Chernobyl
Reactor	Air-cooled graphite pile	RBMK (water-cooled graphite reactor)
Fuel	Natural uranium metal	Low-enriched uranium dioxide
Coolant	Air (Forced air circulation)	Water
Primary use	Plutonium production	Power generation
Cladding	Aluminum	Zircaloy (zirconium alloy)
Key incident	1957 Fire	1986 explosion & fire

In the case of Windscale reactor, aluminum contributed to initial flare-up, particularly in damaged fuel rods.

Some uranium then was oxidized and burned when exposed. Once the fire was initiated, the oxidation of the graphite made some addition to the heat being released and contributed to the development of the fire (Penney, 2017). Flames seen exiting the reactor were composed of carbon monoxide, the oxidation product from graphite when the air supply is limited when the gases emerged into an air-filled cavity. The possibility of a large local release of Wigner energy in a pocket of graphite which had escaped previous annealing was ruled out because the monitoring data indicated that the oxidation rate of graphite was very slow indeed at temperatures below 400 °C. Since no evidence was found for the likelihood of serious 'pocket' releases or of temperatures more than 400 °C, the Committee of Inquiry did not accept graphite "burning" as an explanation of the accident. It is important to emphasize — the graphite did not "burn" in the conventional sense, but it was slowly oxidized by the incoming air. This is a fundamentally different process from combustion — the reaction rate was slow and limited by oxygen supply through the damaged core. An analysis of "what went wrong" is provided in Table 23.

Table 23. A Summary of the Analysis of the Windscale Accident, Pertinent to Graphite Behavior

Step	What went Wrong?
Design	Inadequate understanding of Wigner energy management
Procedure	Faulty annealing procedure led to uneven heating
Monitoring	Core temperatures were not properly controlled
Materials	Aluminum cladding burned; uranium exposed
Graphite	Oxidized (did not combust)
Containment	Air-cooled design allowed radioactive release to the environment

In the case of the Chernobyl accident, during the flawed safety test on April 26, 1986, reactor four experienced a massive power surge (from very low power to 100 times full power in four seconds) this massive increase in energy in the core leading to the initial rupture of the structure and displacing the huge steel cover plate above the graphite stack. This also overheated the fuel rods to the point where the zircaloy cladding (zirconium alloy) began to react with steam, producing hydrogen (IAEA, 1992), (NEA, 2002).

$$Zr + 2H_2O \rightarrow ZrO_2 + 2H_2$$

This is a highly exothermic reaction, adding further heat energy to an already unstable system and generating explosive hydrogen gas. There were then two additional explosions: The first was likely a steam explosion from superheated water flashing to steam. The second was likely a chemical explosion fueled by hydrogen and zirconium reactions. These explosions completed the destruction of the reactor core and the building. This was followed by the release of fuel and structural materials. Fuel fragments, red-hot graphite blocks, and reactor components were scattered onto the roof and surrounding areas, starting fires wherever there was combustible material. Some of the zircaloy cladding continued to react and burn after being ejected. An excellent analysis of the accident sequence is available (BNES, 1987).

After the explosions, the remains of the graphite moderator were exposed to air. The hot graphite began to oxidize (reacting with oxygen), releasing additional heat and contributing to further degradation of the core. However, this was not the kind of intense self-sustaining combustion seen in traditional fires. Importantly, this was not a "graphite fire" in the conventional sense (where the graphite itself would be the primary fuel). It was a slow, oxidative process releasing heat and gases, contributing to aerosol formation and radionuclide transport — but not flames spreading through graphite like wood or paper.

The primary sources of airborne contamination were: (a) fragmented fuel (uranium dioxide particles);

Fission products (^{137}Cs, ^{131}I, ^{90}Sr, etc.), and aerosols formed from hot core materials (cladding, graphite, control rods)**.** Popular accounts often overstate the "burning graphite" idea because smoke and heat release from oxidation resembled a fire, but chemically and physically, this was graphite oxidation, not combustion.

A summary of the role played by the material that led to descriptions of “burning” is given in Table 24.

Table 24. Summary Table of Chernobyl Events

Material	Role
Zircaloy cladding	Initiated the fire sequence by reacting with steam and air, releasing heat and hydrogen.
Fuel	Ejected and fragmented, contributing to contamination but not "burning" as such.
Graphite	Ejected and oxidizing in air due to its high temperature, contributing to heat and acting as a transport medium for radionuclides, but did not undergo "combustion."

We can thus conclude that ‘fire and explosion’ risk when managing graphite from HTGRs will not present any significant hazard, especially given the comprehensive analysis of these past events and the improved understanding of the chemistry and reactor physics involved.

APPENDIX D. SELF-LIMITING RADIONUCLIDE DECAY CHARACTERISTICS IN IRRADIATED GRAPHITE -THEIR EFFECTS IN ASSURING ADEQUATE SAFETY IN LONG-TERM STORAGE

The inherent radioactive decay characteristics of radioactive nuclides can be explored for their role in the long-term storage safety of irradiated graphite. It is the natural character of the radionuclides to exhibit decay over time, which also results in decrease in temperature and the released pressure over time. The decrease in temperature and pressure contribute to any degradation phenomenon of the container materials of construction to be self-limiting. That is, given the degradation phenomenon, such as creep (NRC, 2012) and material loss due to corrosive chemical reactions. Chemical reactions are generally governed by thermodynamics and kinetics, with potential arrest of chemical reactions in systems where temperature and pressure decrease.

The conceptual modeling was performed, to generate the timeline of radiation, temperature, and pressure changes in irradiated graphite waste containers. The model is based on internationally recognized properties, decay data, and typical behavior observed in graphite-moderated reactor decommissioning projects. This is intended for conceptual understanding and is not site-specific.

Model Assumptions and Inputs

Table 25 summarizes the key assumptions and input data sources used to develop the conceptual model.

Table 25. Parameters Used in the Hypothetical Model

Property	Value/Source	Reference
^{60}Co half-life	5.3 years	(IAEA, 2006)
^{3}H half-life	12.3 years	(IAEA, 2006)
^{14}C half-life	5,730 years	(IAEA, 2006)
^{36}Cl half-life	~300,000 years	(IAEA, 2006)
Initial heat load	<1 W/m³	(NDA, 2012)
Graphite heat capacity	690–720 J/kg·K (at room Temperature)	(Mantell, C. L., 1968)
Graphite thermal conductivity	~100 W/m·K (fresh, lower when irradiated)	(Mantell, C. L., 1968)
Radiolysis gas yield (H2, CO, CO_2)	Variable, function of dose & Moisture	(IAEA, 2006)

Modeling Process

The process consisted of analyzing a representative isotope inventory based on typical irradiated graphite from Magnox or RBMK reactors, with applying radioactive decay laws to each isotope. The decay heat was calculated as a function of time. Assuming natural passive cooling, the decrease in temperature was tracked. Initial gas generation from radiolysis based on dose rate and moisture was estimated. Assumed passive venting in early storage (pressure drops after initial buildup). Long-term equilibrium pressure was then estimated in geological disposal (limited gas generation).

Regulatory Context

The modeling process aligns with general expectations found in NRC guidelines for waste classification, storage system design, and radioactive waste management systems. Relevant NRC documents include:

- NUREG-0800 (SRP 11.4) - Solid Waste Management System
- Regulatory Guide 1.143 - Design Guidance for Radioactive Waste Management Systems
- 10 CFR Part 61 - Land Disposal Requirements

Limitations

This model is a simplified conceptual tool. It does not replace a site-specific waste characterization and safety case, which would require precise sampling data, container-specific design features, and detailed thermal, radiological, and chemical analysis using tools like ORIGEN. It is recognized that the data used for the model contains aleatory uncertainty due to inherent natural variability in waste form composition and environmental conditions. Radioactive decay is governed by probabilistic laws, where the half-life represents the time by which 50% of a given radionuclide population is expected to decay. The data used also has epistemic uncertainty due to knowledge gaps. For example, only limited data on very-long-term behavior of materials (e.g., graphite oxidation, long-term corrosion) are available, if at all. Modeling assumptions (e.g., steady-state conditions, homogeneity) also introduce modeling uncertainty.

The NRC encourages probabilistic performance assessments that explicitly account for these uncertainties. This dataset provides deterministic anchors (known trends) that can be used to constrain and calibrate probabilistic models. Uncertainty treatment ensures conservatism in long-term safety assessment - critical for license approvals.

This model relates only to events that happen inside the storage cask due to radionuclide release over time. Issues related to the seeping of potential ground water or moisture (a highly unlikely event in deep geological storage) into the casks and the effect of design basis earthquake events (RG 1.143) (especially repeat earthquakes over the duration of storage) are beyond the scope of this simplified example analysis.

Results and Significance

Figure 8 illustrates the modeled evolution of radiation, temperature, and pressure inside an irradiated graphite waste container over time. This model reflects typical behavior based on international and NRC guidelines for radioactive waste management.

Radiation, temperature, and pressure decrease over time after reactor shutdown, using a logarithmic scale for time to capture the full range from 1 to 10,000 years. This visual helps clearly convey the significant initial decline in all three parameters—crucial for both engineering design and public communication.

The dataset shows significant decreases in: (a) Radiation: From 100 units (at 1 year) to 0.5 units (at 10,000 years); (b) Temperature: From 100 to 10 units; and (c) Pressure: From 100 to 10 units.

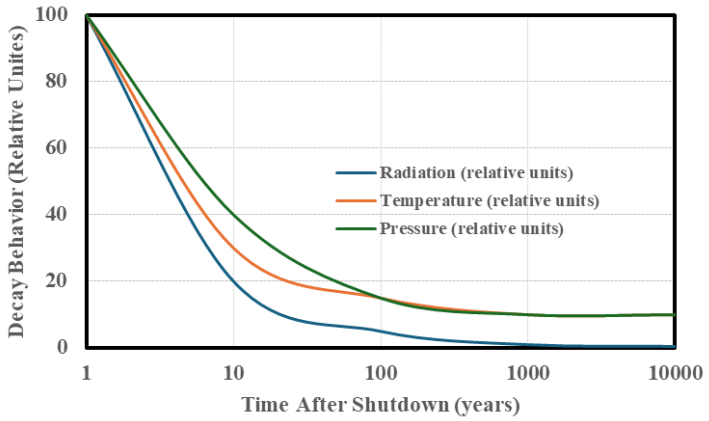

Figure 8. Radiation, Temperature, and Pressure Decay Characteristics Derived from Model Calculation

Table 26. Hypothetical Model Results for Long-Term Reactor-Irradiated Graphite

Feature	**Early Phase (1 – 100 years)**	**Transition Phase (100 – 1,000 years)**	**Long-Term Phase (1,000 – 10,000 + years)**
Radiation	High → moderate	Decreasing	Low/stable
Temperature	High	Decreasing	Low/stable
Pressure	High	Decreasing	Low/stable
Containment need	Very high	Moderate	Long-term integrity
Public risk	Managed by engineering	Declining	Very low
Uncertainty	Highest	Lower	Dominated by long-term material behavior

The data confirms that early containment robustness is essential because high radiation and thermal loads in the first decades demand robust engineered barriers. Regarding long-term performance requirements (e.g., 10 CFR Part 61, 10 CFR Part 63), as conditions stabilize over time, containment design shifts toward chemical and mechanical durability under less aggressive but prolonged conditions. It is highly likely that doses to individuals will not exceed limits (e.g., 100 mrem/year for individual members of the public), and adequate public safety is reasonably assured. This data helps simulate performance of containment and geological barriers to ensure that offsite doses remain acceptably low over time.

A crucial factor is the cooling period after reactors shut down for decommissioning. Allowing the GCC to cool significantly reduces its temperature before transportation and storage. Transportation casks are designed to dissipate heat and maintain safe temperature limits. The cask's thermal conductivity and design play a significant role.

Typically, within 5-years of reactor shutdown, which is an average period for decision to decommission and dismantle a reactor, the GCC cool down to ambient temperatures. This research analyzed the existing data with the results shown in Table 27.

Table 27. Temperature Decay Characteristics of Reactor Core Graphite Subjected to Natural Cooling After Reactor Shutdown

Reactor	**Thermal Power (MWt)**	**Peak Temp Post-Shutdown (°C) (Approximate)**	**Time to Peak Temp (hrs) (Approximate)**	**Temp After 24 hrs (°C) (Approximate)**	**Temp After 1 Year (°C) (Approximate)**	**Temp After 5 Years (°C) (Approximate)**
GA MHTGR (Prismatic) (NRC, 1995)	350	1000	10	< 800	125	65
German AVR (Pebble-bed) (H. Gottaut & Krüger, 1990)	46	1200	5	900	160	75
Chinese HTR-10 (Strydom, 2021)	10	1100	8	850	150	80
Japanese HTTR (JAEA, 2024)	30	1050	6	820	120	70
Chinese HTR-PM (Zhang, et al., 2024)	250 (2x125)	1050	6	830	130	70

All reactors demonstrated effective passive cooling post-shutdown. Peak temperatures occurred within the first 10 hours. By 24 hours, temperatures dropped significantly due to decay heat reduction and natural convection. After 1 year, core temperatures approached near-ambient conditions, depending on containment and insulation. At 5 years, temperatures were within 50-80 °C, highlighting minimal ongoing decay heat. The 2024 experimental data from China's HTR-PM validated predictive models, confirming strong passive heat removal in full-scale operation.

Thus, in Figure 6 the starting temperature is low for any significant chemical reaction to occur, and the likelihood for any reaction to proceed diminishes with time.

The data also supports the typically accepted basic science principles regarding chemical interactions and waste container longevity. As temperature, radiation, and pressure decrease over time, chemical reaction rates between waste and container materials also decrease due to: (a) thermodynamic considerations, such as lower temperatures reducing the Gibbs free energy available for driving corrosive or oxidative reactions; and (b) reduced radiation means fewer radiolysis products that might otherwise promote corrosive chemical

environments. Kinetic considerations are also encouraging. Reaction rates follow the Arrhenius equation

$$k = Ae^{(-(E_a)/RT)},$$

where, A is the preexponential constant, E_a is the activation energy, R is the gas constant, and T is the temperature.

As T drops, reaction rates slow dramatically. Thus, the significance is that at lower long-term temperatures and radiation lessen. Lower temperatures shift equilibrium towards reduced reaction rates and lower nuclear volatility. Container corrosion, graphite oxidation, and gas generation also decrease significantly, supporting the long-term integrity of containment systems, especially important if retrievability is not planned.

With respect to repository design, early high loads may require active cooling, spacing, or specialized shielding. After ~ 1,000 years, passive safety dominates, allowing greater reliance on natural geological barriers.

When comparing the behavior of irradiated graphite in storage with spent fuel in dry cask storge systems, it can be concluded based on prior opinion by NRC (Federal Register, 2020) that scientific "studies and operational experience to date do not preclude a dry cask service life longer than 100 years" that such reasoning would be aptly applicable to the storage of irradiated graphite as well. Additionally, statements such as "the 40-year license period is sufficiently short and the degradation of storage system materials is sufficiently slow that significant storage, handling, and transportation issues are not expected to arise during a single license period, and if information collected during a license period identifies emerging issues and concerns, there would be sufficient time to develop regulatory solutions and incorporate them into future licensing periods (NUREG-2157, Appendix B)" (NRC, 2014). Therefore, "the NRC does not require continuous monitoring" may be true for the storage of irradiated graphite also.

A key focus is on ensuring that irradiated graphite and carbonaceous material in the fuel matrix can be managed safely in interim storage configurations, such as specially designed casks, until final disposal pathways are established. Irradiated graphite can be stored safely in engineered casks, provided that appropriate containment, shielding, and thermal management systems are in place. Graphite irradiated in nuclear reactors contains long-lived radionuclides, including ^{14}C, ^{3}H, ^{36}Cl, and co-deposited metallic fission products, which pose challenges for containment. However, these radionuclides are typically bound within the graphite matrix or deposited in pore spaces, and their release rates are low under stable storage conditions. Because the graphite waste is considered to be GTCC waste, it is bound by NRC's existing regulations, which are likely to be adequate to cover the carbonaceous material in the fuel matrix. In fact, OPTIMUS-L (Optimal Modular Universal Shipping–Light) from NAC International has been formally certified by the NRC under 10 CFR Part 71 as Certificate of Compliance USA/9392/B(U)F-96 for transportation of TRISO-based spent fuel (NRC, 2024). A heavily shielded variant, OPTIMUS-H, is also certified for higher-level radioactive materials. It is explicitly approved to transport high-assay low-enriched uranium TRISO fuel, under the same general CoC as the shielded version, OPTIMUS-H, which also covers spent nuclear fuel.

Several safety assurance factors can be considered as below.

Physical Stability: Irradiated graphite remains structurally stable at ambient temperatures. Although it becomes brittle due to neutron damage, it does not pose a risk of self-ignition or spontaneous degradation in sealed, inert environments.

Gas Generation and Control: While graphite can absorb and subsequently release gases such as tritium, proper cask design—using inert gas backfill (e.g., helium or argon)—can limit gas buildup and corrosion.

Thermal Management: Residual decay heat from graphite is low (especially decades after shutdown), allowing for passive thermal control in storage systems.

Radiation Shielding: Casks can be designed to provide sufficient shielding for beta and gamma radiation from embedded radionuclides. Surface contamination is manageable with standard decontamination and encapsulation methods.

Containment Assurance: Engineered storage systems such as welded steel casks with leak-tight seals can prevent the release of airborne ^{14}C or tritiated water vapor. Additional encapsulation (e.g., coating or overpacking) may be used to enhance containment, especially if graphite is fragmented.

Regulatory Confidence: Existing waste management principles—combined with targeted graphite characterization and packaging strategies—can support regulatory assurance of safe interim storage.

Thus, while irradiated graphite waste is hazardous initially, it becomes significantly less hazardous over time, and the engineered and natural barriers designed to contain it have high reliability.

APPENDIX E. GRAPHITE DUST AND RADIONUCLIDE PLATING IN HTGR AND MSR DESIGNS

Graphite-moderated reactors such as HTGRs and MSRs face unique safety challenges due to radionuclide transport via graphite dust. This Appendix summarizes key lessons from international and historical designs and outlines associated regulatory implications.

Graphite dust deposition leads to both operational and decommissioning safety issues. Operational safety concerns include elevated radiation fields in primary system components, corrosion risks from plated radionuclides, increased worker exposure (dose) during inspection and maintenance, and challenges for early detection of fuel particle failures (Natesan, Purohit, & Tam, 2003). Decommissioning considerations include fixed contamination due to high-temperature plating on nickel-based metal alloys and graphite surfaces, waste classification complications due to embedded fission products, and potential difficulties in decontamination of alloy-graphite interfaces. In MSRs, radionuclides such as noble metals (Ru, Rh, Pd), and tellurium, cesium, iodine, and actinides can plate out from the molten salt carrier onto cooler surfaces in the system. These include off-gas systems, freeze valves, and piping walls. The chemical nature of salt and redox conditions influences deposition behavior (DeGrassi, Nie, & Hofmayer, 2008), (Murphy, 2003). Radionuclide chemical forms may affect leachability and long-term storage safety.

Case Examples and Safety Issues

This research conducted an analysis and evaluation of graphite dust issues. The observations are provided in Table 28.

Table 28. Operational Experience Evaluation and Analysis of Graphite Dust on Graphite Reactors

Reactor	Plating Mechanism	Key Nuclides	Safety Concern	Reference
AVR (Germany)	Graphite dust transport and plating	^{90}Sr, ^{137}C	Dose buildup, hard-to-remove contamination	(Bisplinghoff, Moormann, Nabbi, & Kugeler, 2005)
HTTR (Japan)	Helium-borne particulates	^{110m}Ag, ^{137}Cs	Monitoring challenge	(IAEA, 2006)
HTR-PM (China)	Graphite dust transport and plating	Fission products	Monitoring, design for mitigation	(IAEA, 2006)
Fort St. Vrain (USA)	TRISO breach, graphite erosion	^{137}Cs, ^{90}Sr	Contamination of steam generator	(NRC, 2008)
Peach Bottom (USA)	Dust-assisted radionuclide migration	Mixed fission products	Operational exposure	(Hanson, Engel, & McLaughlin, 1970)
BGRR (USA)	Airborne adsorption	Long-lived	Embedded contamination in	

	onto graphite	isotopes	graphite	
ORNL MSRE (USA)	Volatile radionuclide condensation	Ru, Te, U	Plating in off-gas and drain system piping	(IAEA, 2006)
Magnox (U.K.)	Graphite adsorption, coolant circuit plating	^{60}Co, ^{137}Cs, ^{90}Sr	Decommissioning contamination, worker exposure	(IAEA, 2006)
AGR (U.K.)	Elevated temperature plating and dust	^{14}C, ^{60}Co, ^{110m}Ag	Component activation, maintenance exposure risk	(IAEA, 2006)

NRC Regulations and Guidance

The following NRC regulations and guidance documents are relevant to graphite dust and radionuclide plating issues in both HTGR and MSR systems:

- 10 CFR Part 20 – Standards for Protection Against Radiation, including: Subpart D: Radiation Dose Limits, Subpart F: Surveys and Monitoring
- 10 CFR Part 50 – Domestic Licensing of Production and Utilization Facilities: Appendix A: General Design Criteria, DC 61: Fuel storage and radioactive waste management, GDC 64: Monitoring radioactivity releases
- 10 CFR 50.36 – Technical Specifications
- 10 CFR 20.1402 – Radiological Criteria for License Termination
- 10 CFR Part 70 – Nuclear Material Handling (for MSRs with onsite fuel processing)
- NUREG-1575 – Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)
- NUREG-1757 – Consolidated Decommissioning Guidane
- NUREG/CR-6824 – TRISO Fuel Behavior in HTGRs
- NUREG/CR-6983 – MSR Radiological Source Term and Waste Characteristics
- ORNL-TM-2003/263 – Radiological Lessons from MSRE
- NEI 18-04 – Risk-Informed Performance-Based Guidance for Non-LWRs
- Regulatory Guide 1.233 – Technology-Inclusive, Risk-Informed, and Performance-Based Licensing Guidance

Compliance Requirements for Graphite Dust and Radionuclide Plating Issues

Requirements that reactor designers, operators, and decommissioning planners should consider in ensuring compliance with relevant NRC regulations are provided below, concerning radionuclide plating associated with graphite dust in HTGR and MSR systems.

- Assess and monitor radiation fields in areas with known graphite dust deposition (10 CFR 20,

Subpart F).

- Evaluate design features for control and collection of graphite dust (10 CFR 50, GDC 61 and GDC 64).
- Include graphite dust transport and plating behavior in source term evaluations and design basis accident analyses (10 CFR 50.34).
- Establish technical specifications for radionuclide limits in the primary system and off-gas components (10 CFR 50.36).
- Implement procedures for contamination control and worker protection during maintenance in high-dose areas (10 CFR 20, Subpart D).
- Develop plans for waste classification and disposal of contaminated graphite and plated metallic components (10 CFR Part 61).
- Incorporate graphite dust effects into the final status survey and dose modeling for license termination (10 CFR 20.1402, MARSSIM).
- For MSRs with fuel processing, include plated fission products in material accountability and criticality safety analysis (10 CFR Part 70).
- Apply guidance from NUREG/CR-6824 and NUREG/CR-6983 for expected radionuclide transport and deposition behaviors.
- Follow risk-informed performance-based approaches per NEI 18-04 and RG 1.233 during advanced reactor licensing.

APPENDIX F: INTERIM STORAGE

Introduction

During operation of a HTGR there are mainly two sources of HLW and ILW/LLW stemming from

- Spent fuel being continuously (pebble) or batch-wise discharged (block)
- Exchanged reflector blocks (if this is possible in new designs: every 6-10 years)
- Extracted graphite moderator & absorber pebbles
- Broken fuel fragments
- Dust & captured isotopes from the continuous coolant gas purification

Here, the focus is put on the interim storage of spent fuel. The capacities of the interim storages are usually designed to store the spent fuel at site over the scheduled lifetime of the HTGR plants. External 'Independent Spent Fuel Storage Installation' may also be applied.

Experience in these subjects is mainly provided by Fort Saint Vrain (FSV), which is the only HTGR completely decommissioned, worldwide. The graphite structures of Peach Bottom Unit 1, THTR-300, AVR and DRAGON still remain in their reactor vessels, whereas spent fuel from all these reactors has been removed from the cores and put in sub-critical spent fuel containers being water- or air-cooled within interim storage facilities.

Interim & long-term storage behavior of spent fuel and irradiated graphite components has only been scarcely investigated, documented and reported. This also applies to the rudimentary radiological characterization of spent fuel elements and decommissioned irradiated GCCs.

Most fuel elements from former HTGRs did not contain TRISO particles of high quality, but rather experimental and partially high-enriched BISO or TRISO fuel and mixtures of different fuel types. This applies to DRAGON, AVR, THTR, Peach Bottom Unit 1. Only FSV used TRISO fuel but with a fraction of defective fuel particles exceeding the design targets for actual TRISO fuel developments. Thus, results from former investigations cannot be directly transferred to advanced HTGR designs and TRISO fuel qualities. Current HTGRs under operation (HTTR, HTR-10, HTR-PM) use high-quality TRISO fuel and should provide more representative information once they have reached their targeted burnups.

Spent fuel from molten-salt cooled AHTRs will have to be conditioned and stored differently, with regards to infiltrated salt into the graphite/A3-matrices.

Interim Storage

The German AVR and the interim storage facility of spent AVR fuel were operated at the site of the research centre Juelich (FZJ). Thus, there was an infrastructure available to investigate the storage and disposal behavior in a hot-cell complex and in other radiological laboratories. Currently, the transfer of spent AVR fuel into the interim storage facility in Ahaus is being prepared.

The AVR was primarily used to test the HTGR pebble-bed core concept, the fuel elements themselves, and the components. Fueling of the first core loading started in 1966, with about 30,000 first core fuel elements, 70,000 moderator (graphite) balls, and 3000 absorber balls, in the course of operation, more than 290,000 spherical fuel elements of 15 different types (carbide/oxide, BISO/TRISO, HEU/LEU) with more than 6 billion coated fuel particles plus about 80,000 graphite (moderator) balls were inserted into the core. Fuel-element design also changed during operation from machined graphite shells to pressed matrix materials (Verfondern K., 2006).

There is a significant discharge of graphite moderator and boron- or hafnium-doped graphite absorber pebbles until an equilibrium core configuration has been reached. Similar relations have been accounted for in the THTR, which did not reach the equilibrium core, due to early shut down. These figures show that a considerable amount of moderator or absorbing balls will also be circulated and discharged from pebble-bed

type cores (Dietrich, G. 2024). This compares with the periodically discharged reflector blocks of block-type reactors.

The discharged AVR spent fuel pebbles were first collected in AVR-K canisters containing 50 spent fuel pebbles. The AVR-K canisters were then stored in a water basin for cooling down or directly transferred into larger dry-storage cans for 950 spent fuel pebbles, each (Niephaus, D. 1999).

Additional canisters were used at THTR for the damaged pebbles (diameter 600mm, height 1900mm) and for further debris (diameter 600mm, inner height 1295mm) filled with damaged pebbles and eroded material e.g. from the metallic pneumatic fuel transfer tubing. The debris canister contained about 3 kg of zeolite to reduce humidity and a valve at the bottom to discharge accumulated water.

The relevant gaseous radioactive isotopes for interim storage of irradiated graphite and spent fuel are tritium, ^{85}Kr and ^{14}C, as long as most of the coated-particle fuel is still intact.

Tritium is mainly generated from ^{6}Li impurities in the graphite and A3-matrix by the $^{6}Li(n,\alpha)^{3}H$ reaction, from boron of the absorber pebbles and via the activation of the ^{3}He isotope in the helium coolant by the $^{3}He(n, p)^{3}H$ reaction. The mean tritium content in the matrix of a pebble is in the range of 2×10^{9} Bq. Ternary fission provides about additional 5×10^{8} Bq.

^{85}Kr is mainly generated as a fission product within the TRISO particles but also outside the coated-particles in the A3-matrix and in the pyrolytic carbon (PyC) layers. The free ^{85}Kr is difficult to measure due to the high β & y˜in the irradiated TRISO particles. A numerical evaluation yields about $1.5\text{-}1.75 \times 10^{10}$ Bq per pebble at a burn-up of 18%.

The accumulated ^{14}C activity was about 4.5×10^{7} Bq per pebble. It has higher activity at the surface of the spherical fuel element (Wenzel U. et al. 1979).

Whether such a ^{14}C profile is also expected for other irradiated graphite components in an HTGR core still needs to be validated.

The release of ^{3}H, ^{85}Kr and ^{14}C is governed by diffusion, corrosion and desorption phenomena. In the temperature range for interim storage $30°C < T_s < 140°C$ there is practically no release from intact TRISO particles.

The daily releases of tritium and ^{85}Kr from a fuel pebble in dependence of the temperature are shown in Figure 9. It can be seen that the tritium releases dominate independently of BISO or TRISO fuel against the ^{85}Kr releases, which differ by an order of magnitude between BISO and TRISO fuel.

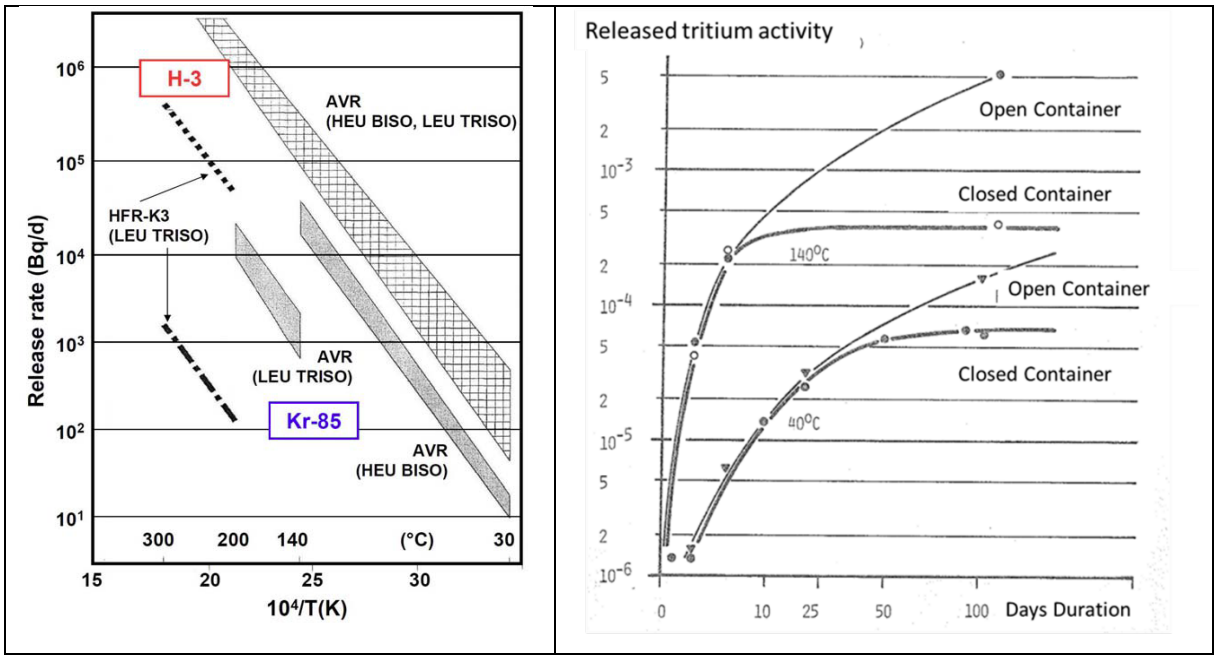

Figure 9: Arrhenius Plot of Experimental Data on Tritium and ^{85}Kr Releases (left) and Tritium Releases from Open Versus Closed Containers [Verfondern K. 2025 (redrawn from Duwe R., 1998)]

The tritium is mainly released as HTO, and humidity in the environment plays a major role (Duwe R., 1998). The access of atmospheric humidity leads to an ongoing release of tritium compared with closed containers. It was recognized that fuel pebbles stored in humid air can absorb 20-40g of (tritiated) water. Thus, spent HTGR should be dried before being hermetically enclosed in interim storage canisters.

The radiocarbon releases also show some temperature dependence, but they are mainly driven by radiolytic corrosion in the strong gamma-field of the spent fuel. Figure 10 shows that the oxygen within a sealed canister is consumed while building up $^{14}CO_2$. This means that the spent fuel from HTGR should preferably be stored under an inert atmosphere.

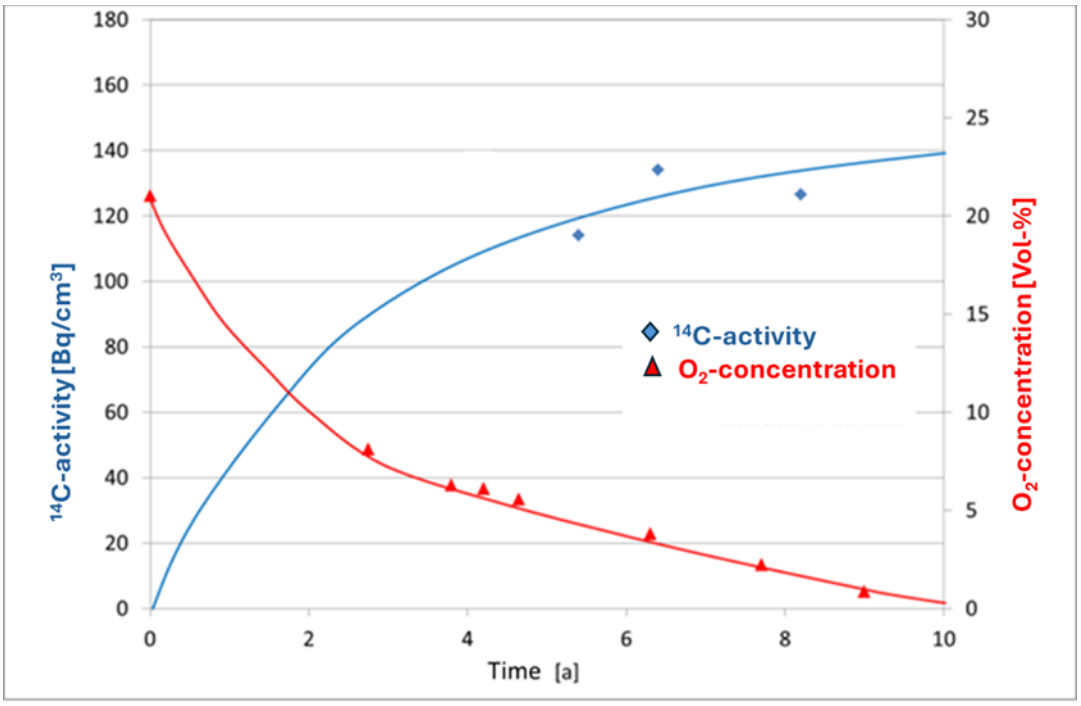

Figure 10: Radiocarbon Release Inside a Sealed Canister [redrawn from: (Duwe R., 1998)]

Figure 11 shows the γ-dose rate profiles measured at the casks' surface after 1 yr of decay time (top) and about 7 yrs later (bottom) [Duwe, 1997]. These results provide some experimental data, which might be of importance for storage and transport issues.

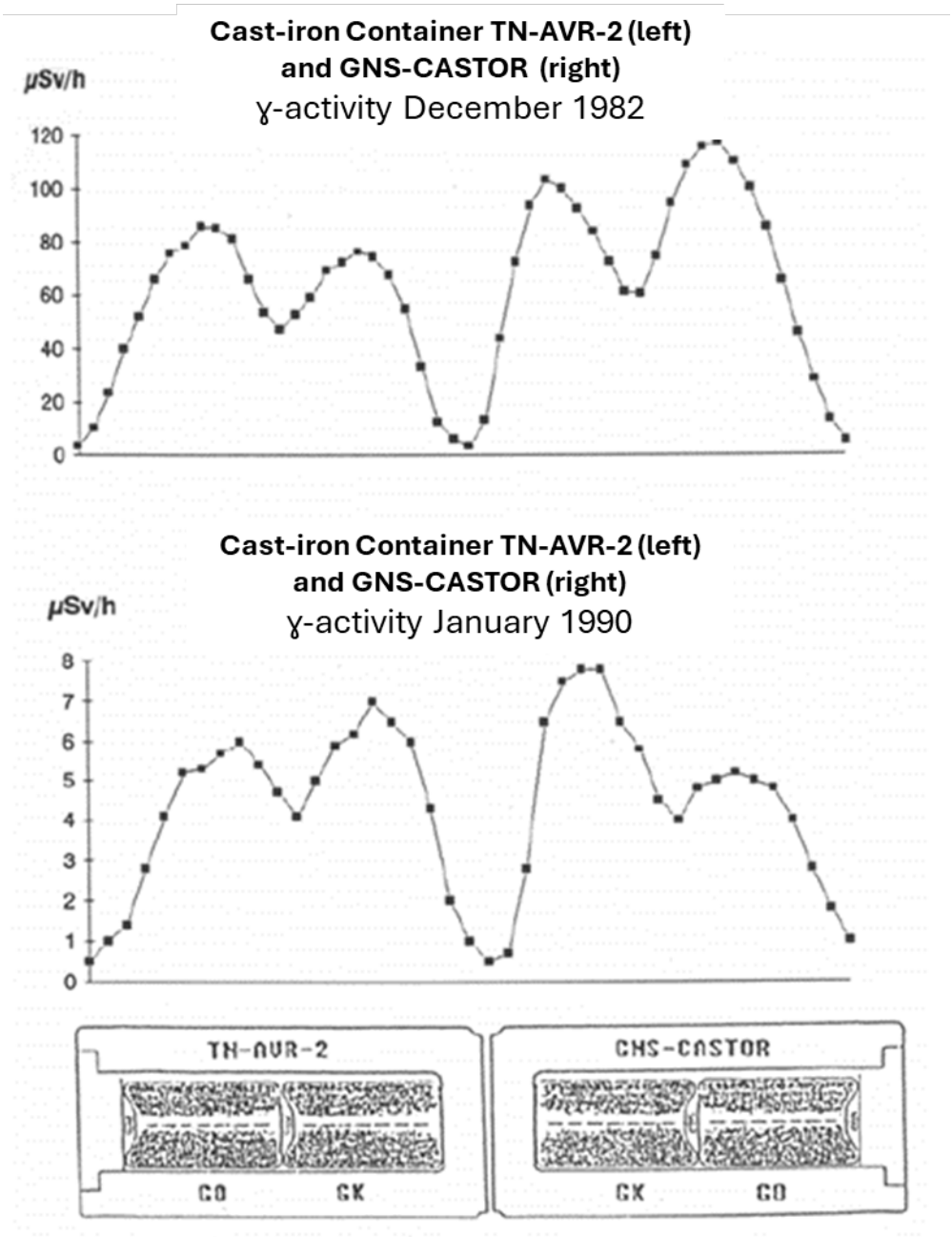

Figure 11: γ-radiation Profiles at the Cask Wall Surface [modified Verfondern, K., 2025 (IAEA, in press)]

APPENDIX G: BAKED CARBON

Introduction

In gas-cooled reactors, there is also the need to apply ceramic refractory materials for thermal insulation purposes. In the past and in present HTGR (HTR-10, HTR-PM), non-graphitized carbon materials were used e.g. for the bottom and peripheral layers of the core structures to reduce the temperature loads of the pressure vessel material. As such materials were and still are also broadly used in many industrial processes, such as steelmaking, the manufacture techniques of conventional carbon bricks e.g. for blast furnaces were directly used also for nuclear applications.

Baked carbon consists of carbonaceous filler material particles such as calcined anthracite coal, residua from blast furnace structures and graphite particles in a size of 1-10 mm and mixed with powder from coke production and coal tar. The anthracite coal and the coke were pre-treated at about 1150°C to expel the volatile fractions in these materials.

The mixture of the above aggregates was stamped by machine-driven but hand-guided hammers into heated steel forms. Heat treatment was done on 1150°C.

This manufacture process was e.g. also applied to the baked-carbon structures of the German AVR reactor. Figure 12 shows the coarse morphology of the baked carbon (cylinder of 6 cm) and of a broken peace is clearly visible.

Figure 22: Baked Carbon Samples from the German AVR Reactor

The low thermal conductivity of this material is evident, when taking it into the hands, as it feels warm, in contrast to nuclear graphite with high thermal conductivity.

It is obvious that this material does not comply with the usual purity requirements for actual nuclear graphite grades. As it was only used in the low neutron flux areas of the reactor higher impurities were accepted in favor of lower cost.

Characterization

As it can be seen in Figure 13, the binder material contains most of the impurities. The white dots are impurities which have been detected by electron microscopy with back-scattering electrons. These impurities mainly consist of oxides of aluminum, silicon and iron, but the easily neutron-activated elements such as cobalt and nickel are always affiliated to these main impurities (G. Pina, 2008). This is not surprising as normal coal tar and powder from industrial coke production has been used as a binder.

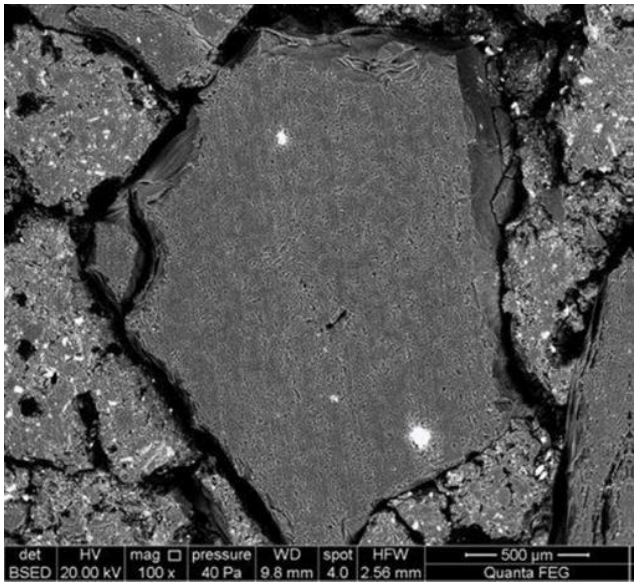

Figure 13 Scanning Electron Micrograph of Carbon Brick in Back-scattering Electron Mode

The consequences of these impurities are listed in Table 29 as main activation products found in the irradiated graphite and in the baked carbon (M. Florjan, 2009)

Table 29: Specific Inventory of Activation Products [Bq/g] in AVR and Comparison with KONRAD Repository Capacity

Nuclide	Graphite (65 Mg)	Baked Carbon (158 Mg)**
^{14}C	7.1 E04	1.8 E06
^{3}H	5.5 E06	7.8 E07
^{60}Co	8.0 E04	8.2 E05
^{36}Cl	2.3 E01	3.7 E02
^{137}Cs	1.0 E04	2.0 E04
^{90}Sr	1.0 E04	5.0 E04

	AVR total inventory	KONRAD max. disposable activity
C 14	2.9 E14	4.0 E14
H 3	3.2 E15	6.0 E17
Co 60	1.4 E14	5.0 E18 **
Cl 36	7.4 E10	
Cs 137 *	8.5 E12	
Sr 90 *	1.9 E13	

* without steam generator
** beta-/gamma nuclides, total

It can be seen that despite the much lower neutron flux in the peripheral and axial (top & bottom) positions all measured activation products are significantly higher in the baked-carbon components. Especially the

extremely high ^{14}C activities will not comply with the WAC for the German KONRAD waste repository and would nearly exhaust the allowed maximum activity of the repository (4×10^{14} Bq) (P. Brennecke, 2015). It is therefore envisaged that irradiated-graphite and baked-carbon components of AVR will be co-disposed with HLW in a new repository, which still needs to be identified and licensed.

In contrast to the former attitudes of taking industrial baked-carbon grades, the ingredients should also be purified according to the requirements for nuclear graphite. Irradiation tests on this kind of carbonaceous material for thermal insulation purposes are not known. It also has a brittle nature and needs to be slowly heated-up and cooled down for avoiding cracks in blocks of larger dimensions.

Irradiated baked-carbon will most probably not comply with LLW categories, due to the immense impurities of former grades even under relative low neutron doses.

References

G. Pina et al., CARBOWASTE Summary Report on Work Package 3, Deliverable 3.0.0, 2008

M. Florjan, PhD-Thesis, RWTH-Aachen, 2009

P. Brennecke; Anforderungen an endzulagernde radioaktive Abfälle (Endlagerungsbedingungen, Stand: Dezember 2014); SE-IB-29/08-REV-2; Salzgitter, 2015