Nuclear Energy University Program

U.S. Department of Energy

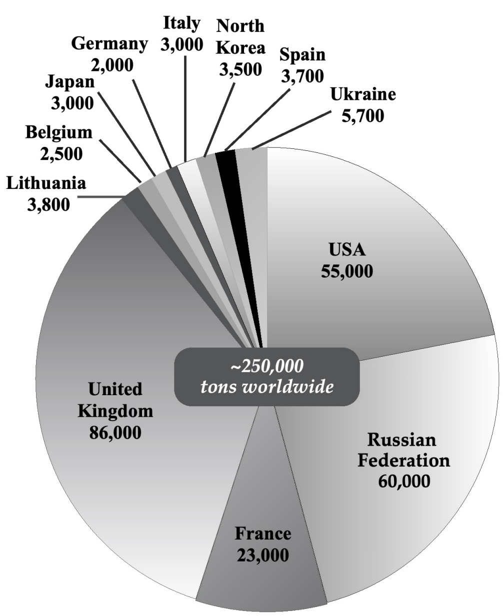
Reduction, Mitigation, and Disposal Strategies for the Graphite Waste of High Temperature Reactors

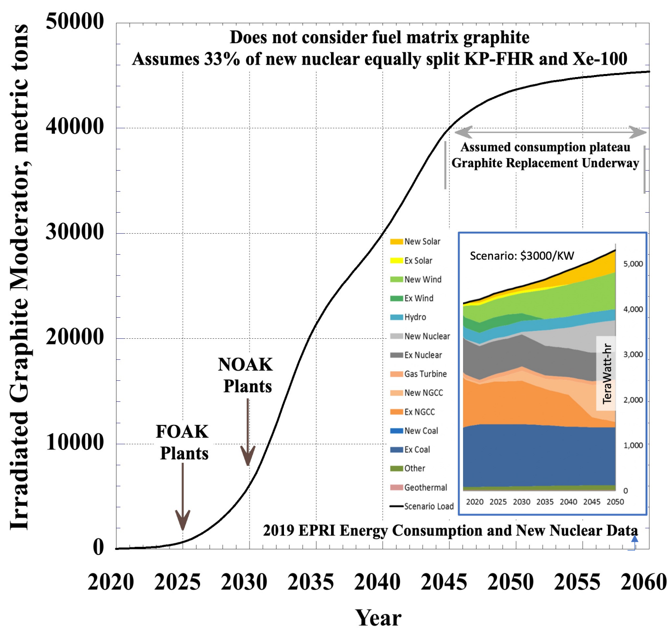
PI: Lance Snead Stony Brook University
Fuel Cycle Research and Development

Co-Leads: Raluca Scarlat UC-Berkeley
Haruko Wainright, MIT

ABSTRACT:

The current waste burden of irradiated graphite, or i-Graphite, is currently estimated to be in excess of 250,000 metric tons worldwide, excepting China.**[1]** The figure below provides a rough international breakdown of the current i-Graphite waste inventory. As a moderator, graphite has been used in a host of early research and power reactors, constituting the vast majority of current inventory. How to deal with this legacy waste burden has been the subject of many decades of research, with early leadership coming from the US and most recently from Europe with a number of comprehensive studies published[1-4]. An important fact regarding i-Graphite is that domestically and internationally there is no regulatory consensus or uniform technology applied to its disposal, with the vast majority of its bulk residing in impermanent vault/silo storage. The recent, significant interest in advanced, high-temperature nuclear to address our climate and energy issues has again raised the issue of i-Graphite waste disposal. This can be quantified is the sheer number of advanced reactor concepts being developed which assume the use of graphite moderator and the significant private and federal investment in such reactors. As specific example, for the combined DOE Advanced Reactor Demonstration and Risk Reduction projects half of the eight designs chosen are graphite moderated. In this context the graphic also provides a simple analysis of the metric tons of i-Graphite that would be realized on the US pathway to advanced nuclear. Within these modest assumptions* the US would soon double its i-Graphite inventory. With this historic and forward perspective, a clear mandate exists to develop solutions to the issue of i-Graphite waste.





**assuming that 33% of the "new nuclear" power as predicted by EPRI will be provided as equal split between Xe-100 and KP-FHR reactors. Analysis does not include fuel matrix graphite.*

As designed, this IRP intends to enable advanced nuclear through development of economically attractive and environmentally sound i-Graphite management strategies resulting in a specific primary objective of quantitative cost savings. This will be achieved through a combined modeling, analysis, technology development, and disposal science and regulatory studies campaign. Our team is comprised of three leading research universities: Stony Brook, UC-Berkeley, and MIT. In Task 1 we will directly involve our vendor partner Kairos Power and X-energy to carry out designs which minimize volume and carry out an in-depth program of characterization and understanding of how ^{14}C is created, moves, and can potentially be operationally mitigated. Reduction of activation products is directly addressed through a partnership with Ibiden to develop a low-nitrogen graphite and the study of very pure grades of graphite. Moving beyond this *source reduction task* we seek to *limit the impact* of the graphite prior to disposal through specific waste mitigation technologies. Here we explore the basics of salt decontamination of i-graphite and waste volume reduction through direct recycle or repurposing i-graphite as an alternative to disposal. In this task (Task 2) we also carry out a systematic determination of the ^{14}C oxidation kinetics required to inform MOOSE for the purpose of billet surface conditioning or waste treatment. Beyond waste mitigation we look toward the final repository science and regulatory issues (Task 3). Here we bring together the domestic and international community in a series of workshops as part of a comprehensive LLW i-graphite framework development program. Key to this work will be to review the 40-year-old scientific basis forming the current ^{14}C maximum allowable concentrations for LLW (as compared to irradiated metal and others) and to investigate how they can be revised by recent scientific evidence and waste isolation technologies. Tasks 1-3 will inform, and be informed by, comprehensive flow-sheeting analysis leading to a technoeconomic analysis supporting our primary objective of a 12.5-25% O&M costs savings.

Primary Objective: To develop a regulatory strategy and portfolio of technologies supporting an environmentally sound and economic framework for low-level waste disposal of i-Graphite.

The following are targeted outcomes of this experimental, modeling, and analytical program:
Outcome 1. Orchestrate a comprehensive cradle-to-grave program mitigating i-Graphite waste burden by: 1) minimizing the volume and activity of i-Graphite produced 2) maximizing the amount of i-Graphite recycled or repurposed prior to disposal, and 3) developing a repository framework for low-level waste.
Outcome 2. To advance specific technologies for handling and treating of i-Graphite waste of both helium and salt-cooled reactors and through combined process, process flowsheet and technoeconomic analysis make informed decisions on next-step technologies.
Outcome 3. Provide the scientific basis and regulatory framework for ^{14}C disposal enabling i-Graphite Low Level Waste disposal consistent with sound economics and environmental policy.
Outcome 4. Maximize workforce development through a university-led, industry and laboratory-integrated research initiative in an area critical to our national energy needs.

References:
[1] IAEA, Processing of Irradiated Graphite to Meet Acceptance Criteria for Waste Disposal: Results of a Coordinated Research Project, IAEA-TECDOC-1790 (2016).
[2] I.A.E. Commission, Treatment and Conditioning of Radioactive Graphite from Decommissioning of Nuclear Reactors, IAEA-TECDOC-1521 (2006).
[3] IAEA, High Temperature Gas Cooled Reactor Fuels and Materials, TECDOC-1645 (2010).
[4] C. Wood, Graphite Decommissioning: Options for Graphite Treatment, Recycling, or Disposal, including discussion of Safety-Realted Issues, EPRI Final Report EPRI 1013091 (March 2006).