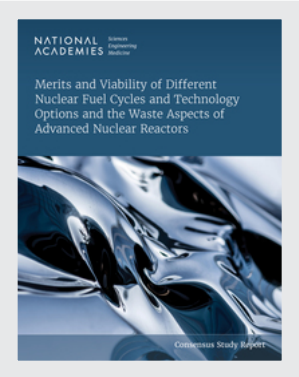


Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Nuclear Reactors (2023)

DETAILS

314 pages | 8.5 x 11 | PAPERBACK

ISBN 978-0-309-29508-6 | DOI 10.17226/26500

CONTRIBUTORS

Committee on Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Nuclear Reactors; Nuclear and Radiation Studies Board; Board on Energy and Environmental Systems; Division on Earth and Life Studies; Division on Engineering and Physical Sciences; National Academies of Sciences, Engineering, and Medicine

SUGGESTED CITATION

National Academies of Sciences, Engineering, and Medicine. 2023. *Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Nuclear Reactors*. Washington, DC: The National Academies Press. https://doi.org/10.17226/26500.

NATIONAL ACADEMIES Sciences Engineering Medicine

NATIONAL ACADEMIES PRESS
Washington, DC

Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Nuclear Reactors

Committee on Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Nuclear Reactors

Nuclear and Radiation Studies Board

Division on Earth and Life Studies

Board on Energy and Environmental Systems

Division on Engineering and Physical Sciences

Consensus Study Report

NATIONAL ACADEMIES PRESS 500 Fifth Street, NW Washington, DC 20001

This activity was supported by contract DE-EP0000026/89243220FNE400048 between the National Academy of Sciences and the U.S. Department of Energy's Office of Nuclear Energy. Any opinions, findings, conclusions, or recommendations expressed in this publication do not necessarily reflect the views of any organization or agency that provided support for the project.

International Standard Book Number-13: 978-0-309-29508-6
International Standard Book Number-10: 0-309-29508-4
Digital Object Identifier: https://doi.org/10.17226/26500

This publication is available from the National Academies Press, 500 Fifth Street, NW, Keck 360, Washington, DC 20001; (800) 624-6242 or (202) 334-3313; http://www.nap.edu.

Printed in the United States of America.

Suggested citation: National Academies of Sciences, Engineering, and Medicine. 2023. *Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Nuclear Reactors*. Washington, DC: The National Academies Press. https://doi.org/10.17226/26500.

The **National Academy of Sciences** was established in 1863 by an Act of Congress, signed by President Lincoln, as a private, nongovernmental institution to advise the nation on issues related to science and technology. Members are elected by their peers for outstanding contributions to research. Dr. Marcia McNutt is president.

The **National Academy of Engineering** was established in 1964 under the charter of the National Academy of Sciences to bring the practices of engineering to advising the nation. Members are elected by their peers for extraordinary contributions to engineering. Dr. John L. Anderson is president.

The **National Academy of Medicine** (formerly the Institute of Medicine) was established in 1970 under the charter of the National Academy of Sciences to advise the nation on medical and health issues. Members are elected by their peers for distinguished contributions to medicine and health. Dr. Victor J. Dzau is president.

The three Academies work together as the **National Academies of Sciences, Engineering, and Medicine** to provide independent, objective analysis and advice to the nation and conduct other activities to solve complex problems and inform public policy decisions. The National Academies also encourage education and research, recognize outstanding contributions to knowledge, and increase public understanding in matters of science, engineering, and medicine.

Learn more about the National Academies of Sciences, Engineering, and Medicine at **www.nationalacademies.org**.

Consensus Study Reports published by the National Academies of Sciences, Engineering, and Medicine document the evidence-based consensus on the study's statement of task by an authoring committee of experts. Reports typically include findings, conclusions, and recommendations based on information gathered by the committee and the committee's deliberations. Each report has been subjected to a rigorous and independent peer-review process and it represents the position of the National Academies on the statement of task.

Proceedings published by the National Academies of Sciences, Engineering, and Medicine chronicle the presentations and discussions at a workshop, symposium, or other event convened by the National Academies. The statements and opinions contained in proceedings are those of the participants and are not endorsed by other participants, the planning committee, or the National Academies.

Rapid Expert Consultations published by the National Academies of Sciences, Engineering, and Medicine are authored by subject-matter experts on narrowly focused topics that can be supported by a body of evidence. The discussions contained in rapid expert consultations are considered those of the authors and do not contain policy recommendations. Rapid expert consultations are reviewed by the institution before release.

For information about other products and activities of the National Academies, please visit www.nationalacademies.org/about/whatwedo.

COMMITTEE ON MERITS AND VIABILITY OF DIFFERENT NUCLEAR FUEL CYCLES AND TECHNOLOGY OPTIONS AND THE WASTE ASPECTS OF ADVANCED NUCLEAR REACTORS

JANICE DUNN LEE (*Chair*), International Atomic Energy Agency (retired), Bethesda, Maryland
PATRICIA A. BAISDEN (*Vice Chair*), Lawrence Livermore National Laboratory (retired), Houston, Texas
RODNEY C. EWING (NAE) (*Vice Chair*), Stanford University, Stanford, California (*until July 21, 2022*)
MARGARET S. Y. CHU (NAE), M.S. Chu and Associates, LLC, New York, New York
PAUL T. DICKMAN, Argonne National Laboratory, Washington, District of Columbia
CRAIG S. HANSEN, Independent Consultant, Clinton, Tennessee
JOHN C. LEE, University of Michigan, Ann Arbor (*until September 16, 2022*)
EDWIN S. LYMAN, Union of Concerned Scientists, Washington, District of Columbia
ALLISON M. MACFARLANE, The University of British Columbia, Vancouver, Canada
ALBERT J. MACHIELS, Electric Power Research Institute (retired)
CHRISTOPHE POINSSOT, Bureau de Recherches Géologiques et Minières (BRGM), Paris, France
JEFFREY D. SEMANCIK, Connecticut Department of Energy and Environmental Protection, Hartford
KEN B. SORENSON, Sandia National Laboratories (retired)
JASMINA L. VUJIC, University of California, Berkeley (*until September 16, 2022*)
NATHALIE A. WALL, University of Florida, Gainesville
HOUSTON G. WOOD, University of Virginia, Charlottesville (*until September 16, 2021*)

Staff

CHARLES D. FERGUSON, Study Director and Senior Board Director, Nuclear and Radiation Studies Board (NRSB) and Board on Chemical Sciences and Technology, Division on Earth and Life Studies (DELS)
OURANIA KOSTI, Senior Program Officer, NRSB, DELS
CATHERINE F. WISE, Program Officer and Co–Study Director (*from July 2022*), Board on Energy and Environmental Systems, Division on Engineering and Physical Sciences
LAURA D. LLANOS, Finance Business Partner
DARLENE GROS, Senior Program Assistant

NUCLEAR AND RADIATION STUDIES BOARD

Acknowledgments

A number of people and organizations contributed to the successful completion of this report. The committee thanks the U.S. Department of Energy, which sponsored the study, and Dr. Erica Bickford, who served ably as the sponsor liaison to the committee.

The committee also thanks the presenters and speakers who gave high-quality presentations during the public meetings, as listed in Appendix B, and the organizations, companies, and agencies represented by the presenters for the information they provided to the committee.

The committee is grateful to the staff of the Nuclear and Radiation Studies Board (NRSB) and the Board on Energy and Environmental Systems (BEES) of the National Academies of Sciences, Engineering, and Medicine for organizing and facilitating this study. Study director Dr. Charles D. Ferguson and staff organized the committee meetings and assisted the committee with collecting the information it needed to write its report. The committee thanks especially Dr. Rania Kosti (NRSB) and Dr. Catherine Wise (BEES), who helped draft and organize the report, as well as Darlene Gros (NRSB), who managed the logistics of the meetings, report review, and publication. These additional National Academies staff assisted with report production: Kasia Kornecki (BEES), Lauren Everett, and Radiah Rose.

Reviewers

This Consensus Study Report was reviewed in draft form by individuals chosen for their diverse perspectives and technical expertise. The purpose of this independent review is to provide candid and critical comments that will assist the National Academies of Sciences, Engineering, and Medicine in making each published report as sound as possible, and to ensure that it meets the institutional standards for quality, objectivity, evidence, and responsiveness to the study charge. The review comments and draft manuscript remain confidential to protect the integrity of the deliberative process.

We thank the following individuals for their review of this report:

TODD ALLEN, University of Michigan
ROBERT A. BARI, Brookhaven National Laboratory
BRIAN BOYER, International Atomic Energy Agency
ROBERT J. BUDNITZ (NAE), Lawrence Berkeley National Laboratory (retired)
MARK DEINERT, Colorado School of Mines
JACK EDLOW, Edlow International
ROBERT T. JUBIN, Oak Ridge National Laboratory (retired)
JOHN KESSLER, J. Kessler and Associates, LLC
WARREN F. "PETE" MILLER, JR. (NAE), Texas A&M University
ANDREW SOWDER, Electric Power Research Institute
PAUL J. TURINSKY (NAE), North Carolina State University
JOHN VIENNA, Pacific Northwest National Laboratory
PAUL P. H. WILSON, University of Wisconsin–Madison

Although the reviewers listed above provided many constructive comments and suggestions, they were not asked to endorse the conclusions or recommendations of this report, nor did they see the final draft before its release. The review of this report was overseen by **DAVID E. DANIEL (NAE)**, The University of Texas at Dallas (president emeritus), and **DAVID W. JOHNSON, JR. (NAE)**, Bell Labs, Lucent Technologies (retired). They were responsible for making certain that an independent examination of this report was carried out in accordance with the standards of the National Academies and that all review comments were carefully considered. Responsibility for the final content rests entirely with the authoring committee and the National Academies.

Contents

Executive Summary

The Further Consolidated Appropriations Act of 2020 (Public Law 116-94) and the Consolidated Appropriations Act of 2021 (Public Law 116-260) mandated that the National Academies of Sciences, Engineering, and Medicine examine the merits and viability of different nuclear fuel cycle options, waste aspects of advanced reactors and their fuel cycles, and nonproliferation and security risks of these technologies.

At the current stage of advanced reactor and associated fuel cycle development in the United States, a vast array of concepts could—if realized—significantly diversify the reactor and fuel cycle technologies used for nuclear power generation in the United States, to a level exceeding the diversity of available technologies for light water reactors. Implementing just a few of the most promising reactor concepts and their associated fuel cycles at a large commercial scale would require substantial government and industry investments well beyond 2050. Most importantly, advanced reactors and their associated fuel cycles would not eliminate the requirement for geologic repositories for some radioactive wastes, because even advanced reactors will require disposal of radioactive fission products. Based on presentations and committee expertise, the committee concluded that the introduction and use of advanced reactors will do little, if anything, to mitigate the need for successful management and disposal of nuclear waste. The present strategy in the United States of a once-through fuel cycle, if completed, could safely dispose of all spent nuclear fuel from commercial power generation, including fission products and all actinide elements, in a system of one or more deep geologic repositories. The following paragraphs present the committee's key conclusions; the full list of its findings and recommendations can be found in this report's Summary.

As a top priority, the committee highlights that Congress will need to establish a single-mission entity with responsibility for managing and disposing of commercial nuclear waste. The entity will need continuity of leadership and funding, as well as a consistent disposal strategy; it will also need high technical and scientific competence and the ability to organize and lead research programs and large construction projects. Importantly, such an entity will need to engage the public in a way that engenders trust. Finally, the entity will need to operate effectively over the many decades that will be required to manage the present inventory of nuclear waste, as well as waste generated by future advanced reactors.

The committee also underscores the importance of the U.S. Department of Energy's (DOE's) decision making about its advanced reactor technology programs. Using data from the Advanced Reactor Demonstration Program and DOE's research and development programs over the next several years, DOE will need to select and support, with industry cost sharing, the development of a *few promising* advanced reactor technologies and fuel cycles, which can be potentially deployed by 2050 and achieve goals described in the Nuclear Energy Innovation Capabili-

ties Act of 2017 (NEICA) (Public Law 115-248). DOE needs to develop a clear and transparent decision-making process based on criteria and metrics that can guide its programs and associated budget decisions. With NEICA's goals as guidance, DOE's criteria will need to include (1) science-based estimates of improved fuel utilization and reduced waste yields compared with the existing light water reactor (LWR) fleet, (2) the development of acceptable waste forms and disposal options, (3) the implementation of enhanced safety throughout the entire fuel cycle similar to that demanded for reactor design and operation, and (4) a level of proliferation resistance comparable to the LWR once-through cycle. DOE will also need to factor into its decision-making process the effort and costs required for establishing advanced fuel cycles, including the manufacturing base and supply chain infrastructure required to support these advanced fuel cycles.

While it will be challenging to make cost estimates with small uncertainties, better understanding of costs for various scenarios of reactor deployment and supporting fuel cycle requirements will aid Congress and DOE in their decision making as to what technologies to support in the coming years. Congress and DOE will benefit from obtaining an independent assessment of cost estimates of various scenarios for potential deployment of advanced reactor technologies and related fuel cycle components. The independent assessor should have expertise in evaluating large-scale construction projects, examining project management challenges, and understanding technological and financial risks and their uncertainties.

Summary

The United States has deployed commercial nuclear power since the 1950s, and as of 2021, nuclear power accounted for approximately 20 percent of U.S. electricity generation. The current commercial nuclear fleet consists entirely of thermal-spectrum, light water reactors (LWRs) operating with low-enriched uranium dioxide fuel in a once-through fuel cycle.[1] In recent years, the U.S. Congress, U.S. Department of Energy (DOE), and private sector have expressed considerable interest in developing and deploying advanced nuclear reactors to augment, and possibly replace, the U.S. operating fleet of reactors, nearly all of which will reach the end of their currently licensed operating lives by 2050. Much of this interest stems from the potential ability of advanced reactors and their associated fuel cycles—as claimed by their designers and developers—to provide a number of advantages, such as improvements in economic competitiveness, reductions in environmental impact via better natural resource utilization and/or lower waste generation, and enhancements in nuclear safety and proliferation resistance.

In the Further Consolidated Appropriations Act of 2020 (Public Law 116-94), Congress directed DOE to contract with the National Academies of Sciences, Engineering, and Medicine to examine two broad issues related to advanced nuclear reactors and nuclear fuel cycles:

(1) merits and viability of different nuclear fuel cycles, including fuel cycles that may use reprocessing, for both existing and advanced reactor technologies;
(2) waste management (including transportation, storage, and disposal options) for advanced reactors, and in particular, the potential impact of advanced reactors and their fuel cycles on waste generation and disposal.

The Consolidated Appropriations Act of 2021 (Public Law 116-260) expanded the role of the National Academies with additional tasking to examine

(3) nonproliferation implications and security risks of fuel cycles for advanced reactors.

In response to these congressional requests, the National Academies assembled a committee of experts (referred to as "the committee" in this report). In addressing the above issues, the committee was tasked with

[1] A thermal reactor primarily uses slow or thermal energy neutrons for fission of the nuclear fuel; a once-through fuel cycle involves a single use of the nuclear fuel, after which the spent fuel is destined for disposal.

focusing on advanced reactors that could be commercially deployed by 2050 and on technologies being investigated by DOE's Office of Nuclear Energy (DOE-NE) and by other relevant programs, such as the Generation IV International Forum and the International Atomic Energy Agency. In December 2021, DOE created the Office of Clean Energy Demonstrations, as mandated by the Infrastructure Investment and Jobs Act (Public Law 117-58); this new office has the primary role in managing the two major advanced reactor demonstration projects in DOE's Advanced Reactor Demonstration Program.[2]

The Nuclear Energy Innovation Capabilities Act of 2017 (NEICA) (Public Law 115-248) defines an *advanced nuclear reactor* as "a nuclear fission reactor with significant improvements over the most recent generation of nuclear fission reactors," where such improvements may include "(1) inherent safety features, (2) lower waste yields, (3) greater fuel utilization, (4) superior reliability, (5) resistance to proliferation, (6) increased thermal efficiency, and (7) the ability to integrate into electric and nonelectric applications."[3] Advanced reactors under development span a broad range of designs.[4]

The advanced reactors currently under various stages of development worldwide differ greatly in their designs and ultimately in their operation. Variations on advanced reactor designs may include (1) different types of coolant besides water, such as liquid metals (e.g., sodium, lead, lead-bismuth), molten chloride or fluoride salts, or helium gas; (2) different neutron energies to induce fission, either thermal neutrons (neutron energy ~0.025 electron volt [eV]) or fast neutrons (neutron energy > ~100 kilo-eV); (3) different fuel forms, such as metals and alloys, TRistructural ISOtropic (TRISO) particles, carbides, and nitrides; (4) different fuel types, such as uranium with 10 percent to just under 20 percent enrichment in uranium-235 (high-assay low-enriched uranium [HALEU]) or thorium to breed fissile uranium-233; and (5) different sizes in terms of generating power, such as microreactors and small modular reactors that would have a smaller plant footprint than large nuclear power plants and a potential for standardized modular factory construction, enhanced safety features, and flexible and incremental power generation. Additionally, the reactors can have either solid, stationary fuels as in conventional LWRs; solid fuels in constant motion, as in pebble-bed reactors; or even liquid fuels dissolved in the coolant and circulated continuously, as in some molten salt reactor designs. Many of the advanced reactor designs are derived from concepts proposed several decades ago, and some have a relatively long history of prior development.

The various advanced reactor design concepts have different fuel cycle options. The U.S. Nuclear Regulatory Commission defines the *nuclear fuel cycle* as the progression of nuclear fuel from creation to disposal, which includes the front-end steps of mining and milling uranium, chemical conversion of uranium, enrichment of uranium, deconversion of the enriched uranium, and fabrication of nuclear fuel for use in reactors; reactor operation; and the back-end steps of spent fuel storage, any reprocessing or recycling operations, transportation of the spent fuel, and final disposal of spent fuel and nuclear waste. The committee examined potential fuel cycles for existing and advanced reactor technologies, including once-through fuel cycles, as well as partially or fully closed fuel cycles that involve reprocessing and recycling.

Most of the advanced reactors being developed in the United States use HALEU and are small modular reactors, which is defined notionally as having a power output of less than 300 MWe (megawatts electric) and is envisioned for factory construction and modular installation. Almost all developers told the committee that they are planning on an open, once-through fuel cycle for at least the near to intermediate terms. Several developers noted that in the longer term, their technologies have the potential for recycling spent nuclear fuel.

---

[2] This sentence was altered to correct the role of the Office of Clean Energy Demonstrations after release of a prepublication version of the report.

[3] This report addresses potential merits of advanced nuclear reactors regarding waste, fuel utilization, and proliferation resistance and focuses the safety assessment on the front and back ends of the fuel cycles. Potential merits related to reliability, thermal efficiency, nonelectricity applications, and reactor operational safety are addressed in the National Academies study Laying the Foundation for New and Advanced Reactors in the United States, which is scheduled to conclude in early 2023. The two studies proceeded independently from one another but had one common committee member. Importantly, the findings and recommendations in each report are the consensus products of each committee acting independently.

[4] While the U.S. Nuclear Regulatory Commission does not define small modular LWRs as advanced reactors, this report does consider them as such.

No country has yet implemented a fully closed fuel cycle with multirecycling of plutonium and partitioning and transmutation of the minor actinides in a fast reactor, although there are potential benefits of doing so: extracting more energy from fissionable materials, greatly reducing or even eliminating the need for enrichment, reducing the mining or importing of additional uranium, and lessening the radiotoxicity and heat load of high-level waste going to a repository. At the same time, significant costs are required for closing the fuel cycle, the most difficult to surmount being the large and sustained investments in fuel cycle infrastructure (reprocessing and fuel fabrication facilities, non–light water advanced reactors, geologic repository needed for all disposal options) that does not currently exist in the United States. Furthermore, for any advanced reactor and associated nuclear fuel cycle to be deployed, additional proliferation, security, and safety risks will need to be addressed and managed. An important, but often overlooked, aspect is the human capital needed—a trained workforce to support advanced reactors and fuel cycles.

As the committee carried out its work, it appreciated that trade-offs are necessary when assessing potential merits and viabilities of different advanced reactors and associated fuels and fuel cycles. Different designs and associated fuel cycles have different potential benefits. However, not one advanced reactor technology can concurrently provide for all the potential benefits relevant to the scope of this study. For example, use of pebble-bed TRISO fuel could enhance safety with its potential for withstanding high temperatures beyond those of current fuels, but it would significantly increase the volume of irradiated graphite waste produced, creating challenges for waste management and disposal. Similarly, closed fuel cycles with continuous recycling could substantially improve utilization of fissionable materials for energy production in a carbon-constrained world—potentially eliminating the need for uranium enrichment. However, a closed fuel cycle would place significant inventories of potentially weapons-usable materials at security risk in reprocessing and fuel fabrication facilities, in reactors, at storage sites, and during transportation operations. These risks could require significant additional resources for international nuclear safeguards, including physical protection systems designed to secure and prevent diversion of these materials. Furthermore, significant research is in progress to address many of the gaps in understanding of the potential benefits and risks of these new technologies. But until advanced reactor fuel cycle concepts go from paper studies and computer-aided design drawings to demonstration and operating units, it is impossible to understand the myriad trade-offs the different design concepts represent and thereby choose a "best in class."

As this study was congressionally mandated, its primary focus is U.S. policy makers and the U.S. nuclear industry. This report and its findings and recommendations provide advice to the U.S. government to help inform the decisions on the development of advanced reactors and fuel cycles in the United States. The following sections present the committee's complete list of findings and recommendations, indicating the topical subject to which they relate. Notably, the findings and recommendations are not presented in order of importance, but rather the order in which they appear in the chapters.

MERITS AND VIABILITY OF THE EXISTING NUCLEAR FUEL CYCLE FOR U.S. LIGHT WATER REACTORS

While the United States has had some experience with commercial reprocessing via monorecycling until the early 1970s, since then it has exclusively deployed the once-through fuel cycle. The relatively low cost of uranium, the abundance of natural uranium, and the deployment of more economical methods for enriching uranium have all contributed to the viability of the once-through fuel cycle for LWRs, not only in the United States but also in most nuclear power–producing countries. Reprocessing of LWR fuel has continued in France and Russia, with varied degrees of commercial success. Presently, the United States has no incentives to undertake monorecycling, largely because of the high costs involved and the decreasing contribution by LWRs to the generation of electricity due to plant shutdowns; substantial challenges (based on past experience) with licensing and construction of spent fuel reprocessing and mixed oxide fuel fabrication installations; security and environmental concerns; and the abundance of natural uranium and uranium enrichment at relatively low costs for the foreseeable future. The clear path for the existing U.S. spent fuel inventory is the once-through nuclear fuel cycle, which is still not being implemented fully because of the political impasse over the Yucca Mountain geologic repository site in Nevada.

Finding 1: Substantial, sustained investments to 2050 and beyond are required to develop technically complex advanced nuclear technologies and fuel cycle facilities and to enable potential commercial success. Notably, France has had a consistent vision on nuclear energy's role in its energy security for more than five decades, and as a result, the majority of France's electricity comes from nuclear power; however, after delaying development of fast reactors, it has yet to close the fuel cycle and is still decades from doing so. No matter what fuel cycle option the United States chooses—whether direct geologic disposal or a closed fuel cycle using advanced technologies—long-term vision and significant and sustained financial commitment will be required to execute it.

Finding 2: Continued use of the once-through fuel cycle for the existing U.S. light water reactor (LWR) fleet has several merits: (1) lower cost compared with any fuel cycle that involves reprocessing and recycling, (2) a reliable international market for nuclear fuel services from multiple suppliers (although that could be disrupted by international crises, such as war)[5]; (3) compatibility with the projected available uranium resources; (4) well-understood proliferation resistance of the entire fuel cycle; and (5) theft resistance of spent nuclear fuel. However, the once-through cycle remains incomplete in the United States because there is still no progress toward establishing an operating geologic repository for the spent fuel from nuclear power plants. Pursuing the monorecycling fuel cycle with existing LWRs in the United States would add cost to nuclear power generation but produce no significant benefits, given the projected abundant supply of natural uranium and uranium enrichment at relatively low cost for the foreseeable future.

POTENTIAL MERITS AND VIABILITY OF ADVANCED REACTORS AND ASSOCIATED FUEL CYCLES

Internationally, several countries are pursuing development and deployment of advanced reactor technologies and associated fuel cycles. The Generation IV International Forum (GIF) involves the United States, the European Atomic Energy Community (Euratom), and more than a dozen other nations pursuing development and deployment of advanced reactor technologies. However, the committee has focused its assessment on U.S.-based efforts. Like GIF, DOE's framework for advanced reactor development emphasizes improved resource utilization, waste minimization, enhanced safety and reliability, and proliferation resistance. In addition, DOE prioritizes versatility more highly compared with earlier generations, especially the ability to provide nonelectrical services, such as desalination, process heat, and hydrogen production. Like GIF designs, DOE's advanced designs have potential improvements that could manifest themselves in a number of ways, such as inherent or passive safety features, simplified or modular designs for ease of fabrication, scalability and enhanced load-following capabilities to complement sources of renewable energy, increased safety of accident-tolerant materials, and fast neutron spectrums for increased fuel utilization via closed fuel cycles.

Most of the advanced reactors being developed in the United States are small modular reactors. In addition to the potential benefits described by DOE, motivations for the development of small modular reactors include smaller plant footprint, potentially lower operation and maintenance costs and lower up-front capital costs (excluding the to-be-expected higher costs of the first-of-a-kind unit), and improved safety. However, because they have yet to be commercially deployed in the United States, these systems have to demonstrate their operational economic competitiveness compared with larger nuclear power plants and nonnuclear energy systems. The non-LWR small modular reactors will also need to demonstrate that they can meet licensing requirements that include safety assessments by the U.S. Nuclear Regulatory Commission.

Finding 3: Government support to help bring advanced reactor technologies to commercial deployment will take substantial financial and technical resources. Specifically, budget limitations will require the U.S. Department

[5] The October 2020 agreement between the U.S. Department of Commerce and Rosatom (the Russian state nuclear energy corporation) allows Russia to continue to export enriched uranium to the United States, but it reduces the proportions from approximately 20 percent of U.S. demand to no higher than 15 percent from 2028 to 2040. The amendment also limits the natural uranium and uranium conversion services from Russia to an amount equivalent to no more than 5 percent of U.S. enrichment demand from 2026 to 2040.

of Energy (DOE) to make difficult decisions about its advanced reactor research and development programs to guarantee support, via industry cost-sharing, for a *few promising* advanced reactor technologies and associated fuel cycle infrastructure in the next several years. If the Advanced Reactor Demonstration Program is funded consistently and fully by both the government and private industry through completion, information such as costs, reliability, project management, and manufacturing feasibility gained from this program will be key to helping DOE in its decision-making process.

Recommendation A: Using data from the Advanced Reactor Demonstration Program and the U.S. Department of Energy's (DOE's) research and development programs over the next several years, DOE should select and support, with industry cost sharing, the development of a *few promising* advanced reactor technologies and fuel cycles that can be potentially deployed by 2050 and achieve goals described in the Nuclear Energy Innovation Capabilities Act of 2017 (NEICA). DOE should develop a clear and transparent decision-making process based on criteria and metrics that can guide its programs and associated budget decisions going forward. With NEICA's goals as guidance, DOE's criteria should include (1) science-based estimates for improved fuel utilization and reduced waste yields compared with the existing light water reactor (LWR) fleet; (2) the development of acceptable waste forms and disposal options; (3) the implementation of enhanced safety throughout the entire fuel cycle, similar to that demanded for reactor design and operation; and (4) a level of proliferation resistance comparable to the LWR once-through cycle. DOE should also factor into its decision-making process the effort required and cost estimates for establishing advanced fuel cycles, including the manufacturing base and supply chain infrastructure required to support them. However, industry will have the primary responsibility for reactors that can be commercially deployed in the U.S. market.

Finding 4: Most of the advanced reactors, especially the non–light water reactors, will confront significant challenges in meeting commercial deployment by 2050. While at least 10 advanced reactor developers currently aim to deploy their technologies by 2050 in the United States,[6] there are no currently operating fueled prototypes of any of these specific advanced reactor designs in the United States; there are, however, some demonstration and commercial units of similar reactor designs in operation internationally. Moreover, the vast majority of advanced reactors are still in the early design phase. Depending on the maturity of the technology, advanced reactor developers face a range of challenges to bringing the proposed technologies to commercialization, including little or no direct operational experience of some designs at engineering scale; the lack of adequate capabilities to develop, test, and qualify advanced fuels and materials; and as a result, the potential considerable time for regulatory approval.

Recommendation B: To support the development and deployment of advanced reactor technologies, Congress and the U.S. Department of Energy (DOE) need to provide or ensure access to materials testing and fuel qualification capabilities essential to advancing these technologies. Accomplishing this requires a coordinated plan involving DOE's Office of Nuclear Energy, Office of Science, and domestic and international user communities. The plan should consider a full range of alternatives in meeting both short- and long-term needs.

Finding 5: Of the advanced nuclear reactor technologies currently in development, small modular reactors based on light water reactor (LWR) technologies are furthest along toward being connected to the electrical grid. This is because they can leverage the existing LWR and fuel cycle infrastructure and because these technologies have received government and private-investor financial support for more than a decade.

Finding 6: The common perception that the thorium-232/uranium-233 fuel cycle will generate less plutonium and minor actinides (therefore reducing the radioactive hazard of its spent fuel compared with that from the

[6] As of January 2022, this number of developers had submitted applications or preapplications to the U.S. Nuclear Regulatory Commission.

uranium-235/plutonium fuel cycle) is incorrect. Overall, because of the decay of associated actinide products, thorium-based fuels have short- and long-term radiotoxicities (hazards) comparable to uranium-based fuels.

NUCLEAR FUEL CYCLE NEEDS FOR ADVANCED REACTORS

To evaluate the merits and viability of different fuel cycle options, the committee analyzed (1) the once-through cycle for LWRs, (2) monorecycling of uranium and plutonium as mixed oxide fuel in LWRs, and (3) multirecycling of plutonium and minor actinides (americium, curium, and neptunium) in fast reactors.

The committee also analyzed fuel production and fabrication for different fuels that could use HALEU and thorium, and different fuel forms, including TRISO, oxide, pure metal, metallic alloys, nitride, carbide, and liquid fuel salt. Particular emphasis was placed on the supply of HALEU, which almost all of the advanced reactor developers plan to use and for which there is currently no commercial-scale production in the United States. Currently, the only commercial supplier of HALEU is Russia, and as this report was being completed in April 2022, the committee notes that U.S.-based advanced reactor developers' access to that supply might be prohibited, depending on potential U.S. sanctions on Russia due to the Russian war against Ukraine. For each fuel cycle option, the committee considered resource utilization, waste management, proliferation resistance, security risks, and safety aspects (excluding reactor operational safety). Regardless of the fuel cycle, a geologic repository will be required to dispose of spent nuclear fuel, high-level wastes, and other wastes that contain long-lived radionuclides generated by reactor and fuel cycle operations.

Finding 7: There is no current domestic capacity to supply high-assay low-enriched uranium (HALEU) to meet the projected needs of U.S.-based advanced reactor developers over the next decade. Therefore, if reactor projects requiring HALEU continue to advance, identifying a reliable supply of the material will be crucial. Otherwise, many developers will likely initially acquire HALEU from foreign sources, such as Russia, raising concern about ensuring reliable supply. Reliance on foreign sources of HALEU or HALEU feedstock (as many advanced reactor developers had planned to do prior to the invasion of Ukraine by Russia) without a reliable domestic supply could have serious energy and national security implications if advanced reactors using HALEU are adopted widely.

Recommendation C: Given the uncertainty of foreign supply arrangements of high-assay low-enriched uranium (HALEU) for advanced reactors, the U.S. Department of Energy should prepare contingency plans that may include (1) scheduled delays in the development, demonstration, and deployment of these systems; (2) a schedule for industry as to when and what level of federal support will be available; and (3) the release of stockpiles of highly enriched uranium for downblending until domestic and secure supplies are available.

Finding 8: For a nuclear fuel cycle supporting any reactor technology to be viable, it has to be *industrially sustainable*. Although many fuel cycle options are possible, most differ dramatically from the current situation in the United States—the once-through fuel cycle. All elements of a sustainable nuclear fuel cycle would have to be fully demonstrated both individually and together, because what works with computer-aided designs would not necessarily translate to industrial-scale deployment. For that reason, an evolutionary, progressive approach is likely more practical than a revolutionary approach that attempts to solve all potential issues at the same time with advanced technologies. The evolutionary approach is more important for commercial deployability and will require the majority of investment efforts; nonetheless, some investments in high-risk, high-reward approaches may be worth pursuing. The committee agrees with the 1996 National Research Council report *Nuclear Wastes: Technologies for Separations and Transmutation*, which states that advanced fuel cycles will require substantial investment and take between many decades to more than a century of continuous recycling using a separations and transmutation system of appropriate scale, in order to potentially achieve the full benefit of plutonium recycling and partitioning and transmutation of minor actinides.

Recommendation D: The current U.S. policy of using a once-through fuel cycle with the direct disposal of commercial spent nuclear fuel into a repository should continue for the foreseeable future.

The once-through fuel cycle is the baseline, and any new fuel cycles should have advantages over that baseline for them to be deployed. However, so as not to preclude these options in the future, the U.S. Department of Energy (DOE) should continue fundamental studies to evaluate the feasibility of using recycling and transmutation for closing fuel cycles. Specifically, DOE should develop and implement a phased, long-range research and development program that focuses on advanced separations and transmutations technologies.

Finding 9: As proposed for some advanced reactor closed fuel cycles, reprocessing and recycling of spent nuclear fuel introduces additional safety and environmental considerations over the management of open-cycle light water reactor oxide fuels. In assessing the safety and environmental performance of advanced reactors, the risks and environmental impacts will require optimization over the entire fuel cycle, including front-end processes (mining, enrichment, and fabrication), back-end processes (reprocessing and recycling together), and disposal (interim and final). Currently, advanced reactor developers focus primarily on the safety aspects of the reactor and its operation, and put less priority on the safety aspects of other parts of the fuel cycles.

Recommendation E: Congress and the U.S. Department of Energy should incentivize safety improvements across the supporting fuel cycle.

Finding 10: Because of the absence of current commercial operational experience with advanced reactor technologies in the United States, reliable cost data and estimates for these technologies and their associated fuel cycle components are lacking. The costs of advanced reactors and their associated fuel cycles could range from at least several billion dollars—for pilot-scale non–light water advanced reactors and their fuel cycle facilities—to hundreds of billions of dollars—for full deployment of an alternative fuel cycle that would replace the existing once-through cycle and existing light water reactors. Congress and the U.S. Department of Energy will need better understanding of the cost estimates for various scenarios of reactor deployment and supporting fuel cycle requirements to aid their decision making as to what technologies to support in the coming years.

Recommendation F: Congress and the U.S. Department of Energy should obtain an independent assessment of cost estimates of various scenarios for potential deployment of advanced reactor technologies and related fuel cycle components. The independent assessor should have expertise in evaluating large-scale construction projects; examining project management challenges; and understanding technological and financial risks, as well as their uncertainties.

MANAGEMENT AND DISPOSAL OF NUCLEAR WASTES ASSOCIATED WITH ADVANCED REACTORS AND FUEL CYCLES

For its analysis of waste generation from advanced reactors, the committee compared waste issues for four representative advanced reactors—integral pressurized water reactors, high-temperature gas-cooled reactors, sodium-cooled fast reactors, and molten salt reactors—with the once-through cycle for LWRs, which was used as the reference case. Wastes considered were from front-end processes, reactor operations, spent fuel and high-level waste, any processing and/or reprocessing, and decommissioning. The committee also considered the impact of advanced reactors and fuel cycles on geologic disposal.

Finding 11: As the United States nears the 40th anniversary of the Nuclear Waste Policy Act (NWPA) (Public Law 97-425) and its Amendments (Public Law 100-203, Part E), there is no clear path forward for the siting, licensing, and construction of a geologic repository for the disposal of highly radioactive waste (mainly commercial spent nuclear fuel). The United States finds itself in this difficult situation for many reasons, including (1) changes to the original NWPA of 1982 that moved the process of site selection from a consideration of multiple sites to a single site, Yucca Mountain, Nevada; (2) a slowly developing and changing regulatory framework that provided late guidance in the site selection process and the evaluation and comparison of multiple sites; (3) ineffective management

of the Nuclear Waste Fund ($45 billion) by Congress, which treated what was to have been a ratepayer escrow account as if it were taxpayer monies; (4) consequential policy changes occurring with changing administrations; (5) conflicting congressional and executive policies; and (6) insufficient public engagement in decisions concerning the basic strategy for the storage and disposal of the waste. The continued delay in planning and progress has only made the situation more complicated, as the present legal and regulatory frameworks have become outdated and even more limiting. Numerous assessments during the past decade, notably the *Blue Ribbon Commission on America's Nuclear Future* (2012) and *Reset of America's Nuclear Waste Management: Strategy and Policy* (2018), have outlined a way forward. The committee agrees with common recommendations of these studies to establish a single-mission nuclear waste management and disposal entity, for which models have been proposed that deserve consideration by Congress. The entity could be governmental, partially governmental, or private; as to the latter option, the committee notes that two successful programs are being led by fully private entities: Posiva in Finland and SKB in Sweden. Important attributes of the entity are described in Recommendation G.

Recommendation G: Congress should establish a single-mission entity with responsibility for the management and disposal of nuclear wastes.

- **Such an entity should be responsible for "cradle-to-grave" care and disposition of spent nuclear fuel—that is, from its discharge from a reactor plant to its final disposal in a repository. This entity should have continuity of leadership and funding, as well as a consistent disposal strategy. It should also have high technical and scientific competence, be able to organize and lead research programs and large construction projects, and, importantly, be able to engage the public in a way that engenders trust. Finally, the entity should operate effectively over the many decades that will be required to manage the present inventory of nuclear waste, as well as waste generated by future advanced reactors.**
- **Congress should ensure that funds collected from ratepayers that use electricity from nuclear power plants, now more than $45 billion, are applied to the disposal of the spent fuel generated by nuclear power plants and that collection of funds from all commercial generators of nuclear power resumes. Moreover, funding for the entity should be held in a true escrow account and not be subject to the annual appropriations process.**
- **The entity should immediately initiate steps to begin the process of site selection. Before sites are considered, a decision-making process with appropriate technical criteria and an acceptable method of public engagement, such as consent-based siting, needs to be defined in collaboration with impacted communities, tribes, and states. Congress should make a decision on what to do with Yucca Mountain, which could include keeping it as a possible site for consideration, depending on the plans of the new entity.**

Finding 12: The advanced reactor developers' presentations to the committee focused on the reactors themselves, with little or no attention to nuclear waste management or disposal of the nuclear waste generated because there is no incentive for them to do so. In the absence of a final geologic disposal strategy in the United States, the expansion of nuclear power using advanced reactors will add to the amount of spent nuclear fuel and associated waste that requires disposal and increase the complexity of this challenge because of the need to dispose of new types of fuels and waste streams.

Finding 13: Presently proposed advanced reactor technologies will initially use a once-through fuel cycle; however, compared with those currently in use, the fuels will have a higher uranium enrichment (e.g., high-assay low-enriched uranium [HALEU]) and a higher burnup; also, they will use new types of fuel materials and designs (e.g., TRistructural ISOtropic [TRISO] fuels). As compared with the disposal of the present uranium oxide spent fuel, these new fuel types may result in changes of (1) the amounts (either in mass or volume), chemical compositions, and radionuclide inventories of the waste to be disposed; (2) the thermal power of fuel assemblies; and (3) the durability of the spent fuel in a disposal environment. More specifically, from the waste management and disposal perspective, it is important to note the following:

- Radiological risks from disposed waste are dominated by the mobility of long-lived radionuclides and not by the radiotoxicity inventory. Therefore, radiotoxicity itself is a poor metric for repository performance and risk to the public from waste disposal. The long-term safety of disposal of actinides in appropriate geologic settings is largely independent of the actinide inventory of the repository, except in the off-normal situation where the geological barrier is bypassed—for instance, by human intrusion. Because the amount of mobile long-lived fission products generated is independent of reactor type, most advanced reactor technologies will have little impact on estimates of long-term repository performance. Key factors for long-term repository performance are the redox conditions of the geochemical environment, waste form stability, groundwater flow rates, and solubility/sorption of radionuclides. A reducing environment is preferred. Advanced reactor technologies will have little or no impact on these factors.
- The total quantities of fission products generated are generally related to fission rate and are largely independent of reactor technologies, although the distributions of different isotopes may differ. Both short- and long-lived fission products are important on the timescales relevant to geologic disposal. Short-lived fission products (e.g., strontium-90 and cesium-137) produce significant heat, while long-lived fission products (e.g., iodine-129 and technetium-99) are extremely mobile in a repository environment. Advanced reactor technologies will, in general, generate a higher amount of fission products in each spent nuclear fuel package because of their higher burnups, resulting in a higher thermal load. Increased thermal loads of waste containers will impact a number of repository design features, such as the size and spacing of waste packages, the size of the repository footprint, and engineering designs, thereby impacting the cost of repository construction.
- Enhanced stability and durability of waste forms in a repository environment can be beneficial to the performance of a repository by limiting the release of radionuclides from the spent fuel. Some advanced reactor technologies propose using advanced fuel designs with the potential to contain radionuclides (e.g., TRISO fuel), but this potential must first be demonstrated by experimental programs that examine the fuel's long-term integrity in intense radiation fields and at high temperatures.

Recommendation H: The implementer of the nuclear waste management and disposal program, in collaboration with advanced reactor developers, should support research and development on (1) spent fuels from advanced reactors to understand their degradation behaviors in a variety of geologic environments, (2) recycling and reuse options for irradiated graphite, and (3) management and disposal of unique waste streams from advanced reactors that may pose a challenge for geologic disposal. Moreover, the wastes and treatment technologies should be characterized and quantified.

Recommendation I: The principal agencies (U.S. Department of Energy, U.S. Nuclear Regulatory Commission, and U.S. Environmental Protection Agency) should initiate a coordinated effort to develop regulations and standards for a generic repository (i.e., not specific to Yucca Mountain) and new types of spent fuel and waste forms in order to support geologic disposal of new fuel types from advanced reactors. Developers of advanced nuclear reactors also need to anticipate the impact of new fuel types on their performance as a waste form in a geologic repository.

Finding 14: Conceptually, advanced reactors could be used to reduce the current inventory of transuranics in the approximately 86,000 tonnes of legacy spent fuel to date; this would require considerable resources and time to design, develop, prototype, build, and make operational the required infrastructure. Creating this infrastructure is not practicable in the near future, as long as uranium and enrichment services are readily available.

Recommendation J: The immediate-future focus of the U.S. nuclear waste management and disposal program should be planning for the geologic disposal of the existing spent fuel that is presently stored at 79 sites in 35 states and the approximately 2,000 metric tons per year being generated by existing commercial reactors.

Finding 15: Most of the advanced reactor types proposed would generate waste streams for which there is little experience or mature technical ability to manage. All additional waste treatment options would entail additional costs not encountered in the management and disposal of spent light water reactor (LWR) fuel. High-temperature gas reactors will produce much larger volumes of spent fuel compared with equivalent energy production from LWRs. It may be possible to reduce the volume by removing graphite from the spent fuel, but those technologies are immature. Dust production from pebble-bed reactors would pose waste and decommissioning challenges. Sodium-cooled fast reactors would produce large volumes of irradiated sodium waste that would require treatment and disposal; sodium-bonded spent fuel is not suitable for direct disposal and would require treatment by methods not yet technically mature at the industrial scale. Molten salt reactors produce two waste streams, radioactive off-gases and the spent fuel salt waste, that would require processing into waste forms suitable for disposal. These treatment methods and suitable wastes forms are in early stages of exploration. Most of these advanced reactors would produce large quantities of irradiated graphite waste—from use as moderators or reflectors—and this material would prove challenging to manage as well. While European researchers have analyzed graphite waste disposal extensively, researchers in the United States generally lack this expertise.

STORAGE AND TRANSPORTATION

The committee also considered storage and transportation requirements for advanced reactor fuels and materials, including HALEU, TRISO and graphite materials, metallic fuels and materials, and molten salt liquid fuels.

Finding 16: Similar to issues with waste management, advanced reactor developers have not adequately examined the back-end operational management (i.e., storage and transportation) of advanced nuclear spent fuel. Consequently, the stability of waste forms and potential issues related to needed processing prior to storage, as well as repackaging that may be required for transportation and final disposal, have not been studied sufficiently.

Finding 17: Secondary waste streams—such as lead, sodium, molten salts, and irradiated graphite (moderators and/or from TRistructural ISOtropic [TRISO]–particle fuel disassembly)—from advanced reactor and fuel cycle operations will need to be stabilized and packaged for storage prior to downstream operations to support disposal. Waste forms for these secondary wastes can be developed to be compatible with storage regulations by the U.S. Nuclear Regulatory Commission; however, some still require research and development to properly characterize performance envelopes.

Recommendation K: The U.S. Department of Energy (DOE) should require advanced reactor developers that receive DOE funding to work with designers of storage and transportation concepts to mitigate potential fuel cycle disconnects caused by suboptimized designs that satisfy only one operational aspect of the back end of the fuel cycle (e.g., storage, transportation, or disposal). Through venues such as the Extended Storage Collaboration Program of the Electric Power Research Institute, DOE should continue to collaborate with industry to identify and address long-term storage packaging issues and how they may potentially impact downstream transportation and disposal operations. This recommendation applies to all wastes generated from reactor operations and potential reprocessing operations. When appropriate, DOE should consider funding research and development to address common waste form degradation issues and their impact on storage and transportation system designs. The implementer of the nuclear waste management and disposal program should execute this recommendation.

Finding 18: Because of the higher enrichments of fresh high-assay low-enriched uranium (HALEU) and potential higher burnups of irradiated HALEU fuels, maintaining subcriticality margins and having adequate thermal and shielding protection during transport and storage would most likely require at least some of the following:

- criticality experiments for enrichments above 5 percent to support benchmarking analyses;
- assessment of the feasibility of using type 30B containers for transport of enriched uranium hexafluoride, if needed; and
- criticality, thermal, and shielding assessments for storage and transportation.

Recommendation L: In its advanced reactor programs, the U.S. Department of Energy should support funding and provide technical resources for integration of high-assay low-enriched uranium (HALEU) products into advanced reactor fuel cycles by performing criticality, thermal, and shielding assessments of storage and transportation systems to meet stated schedules of deployment for demonstration and prototyping of advanced reactors.

NONPROLIFERATION AND SECURITY RISKS

Deployment of advanced reactors and their supporting fuel cycles will involve the production, transportation, storage, and irradiation of nuclear materials with characteristics that differ significantly from the current U.S. LWR fleet fueled with low-enriched uranium. The committee's evaluation of nonproliferation implications and security risks of fuel cycles for advanced reactors also used the once-through cycle for LWRs as a reference case. For this analysis, the committee grouped advanced reactors and fuel cycles into the following categories: (1) once-through fast reactors using HALEU; (2) pebble-bed reactors using HALEU; (3) once-through molten salt–fueled reactors using HALEU; (4) use of thorium and uranium-233; and (5) closed fuel cycles.

Finding 19: Expanding the global use of high-assay low-enriched uranium (HALEU) would potentially exacerbate proliferation and security risks because of the potentially greater attractiveness of this material for nuclear weapons compared with the low-enriched uranium used in light water reactors. The increased number of sites using and states producing this material could provide more opportunity for diversion by state or nonstate actors.

Recommendation M: The U.S. National Nuclear Security Administration, in coordination with the U.S. Department of Energy's Office of Nuclear Energy, should assess proliferation and security risks associated with high-assay low-enriched uranium (HALEU) and its potential for expanded global use. In parallel, the U.S. government should foster an international effort, which could be facilitated by the International Atomic Energy Agency, to examine and address these risks.

Finding 20: All of the advanced reactor fuel cycles will require rigorous measures for safeguards and security commensurate with the potential risks they pose. Issues requiring special attention include the following:

- Material accountancy (i.e., tracking and quantification) is more difficult for molten salt and pebble-bed technologies than for reactor systems that use stationary solid fuels because of the technical challenges in performing measurements with online fuel and bulk-handling facilities. Containment and surveillance will also be more challenging to implement for these types of reactors. Thorium/uranium-233 fuel cycles require development of safeguards technology because of the large number of variants in their systems. Moreover, safeguards tailored to traditional uranium/plutonium fuel cycles are not applicable to these systems.
- Fuel cycles involving reprocessing and separation of fissile material that could be weapons usable pose greater proliferation and terrorism risks than the once-through uranium fuel cycle with direct disposal of spent fuel, as the separated fissile material would not be uniformly mixed with highly radioactive fission products. Separated, potentially weapons-usable materials could include fissionable materials other than the "traditional" special nuclear materials of highly enriched uranium, plutonium, and uranium-233. Thus, for these closed fuel cycles, specific safeguard technologies will likely be required to meet the International Atomic Energy Agency's goal of timely detection.

Recommendation N: The U.S. government should support the International Atomic Energy Agency's (IAEA's) development and application of effective safeguards for advanced reactor technologies by authorizing, via the U.S. interagency process, IAEA access through the eligible facilities list, especially to those advanced reactor systems for which the IAEA does not currently have safeguards experience.

Developers of these types of advanced reactors and fuel cycle facilities should provide facility information to the IAEA to help with integration of safeguards considerations into the design process.

Recommendation O: The U.S. Nuclear Regulatory Commission should initiate a rulemaking to address the security and material accounting measures for high-assay low-enriched uranium (HALEU) and other attractive nuclear materials that may be present in advanced reactor fuel cycles.

1

Background and Study Task

1.1 MOTIVATION AND REQUEST FOR THE STUDY

In recent years, Congress, the U.S. Department of Energy (DOE), and the private sector have expressed considerable interest in developing and deploying advanced nuclear reactors to augment, and eventually replace, the operating fleet of large light water reactors (LWRs). Much of this interest stems from the potential for advanced reactors and their associated fuel cycles to better support the energy security and low-carbon electricity generation benefits of nuclear energy than existing LWRs because of their abilities, as claimed by their designers and developers, to reduce environmental impact (e.g., via better natural resource utilization, lower waste generation[1]); to provide for safer and more proliferation-resistant nuclear energy systems; to increase the economic competitiveness of nuclear energy generation technologies; and, in some cases, to provide energy applications beyond electricity generation (e.g., process heat, desalination). The projected net loss of nuclear generating power in the United States by the middle of this century provides additional motivation for pursuing new and advanced reactor designs. Of the 95 licensed U.S. LWRs operational in mid-2020 (when this study began), five are planned to close by mid-2025, and all will shut down by 2055 if the current licenses are not extended to allow for 80 years of operational life per reactor (Holt, 2021b). As of the end of 2021, 93 LWRs are operational. Almost all LWRs have received license extensions to permit 60 years of operations; several have recently applied and one has been approved for a license extension to 80 years. Only two new LWRs are under construction, both at the Vogtle Electric Generating Plant in Georgia, and these have had delayed start-ups (originally planned for 2016 and 2017)—with one unit expected to be operational in the first quarter of 2023 and the other by the fourth quarter of 2023 (based on information from August 2022; NEI, 2022).

In the Further Consolidated Appropriations Act of 2020 (Public Law 116-94) and the Consolidated Appropriations Act of 2021 (Public Law 116-260), Congress directed DOE to contract with the National Academies of Sciences, Engineering, and Medicine to evaluate these claims, with particular consideration to fuel cycles, waste management, and nonproliferation. The committee's statement of task is shown in Sidebar 1.1. In response to these congressional requests, the National Academies assembled a committee of experts (referred to as "the committee" in this report) tasked with focusing on those advanced reactors that could be commercially deployed by 2050 and

[1] *Resource utilization* is typically measured as the amount of natural uranium or thorium fuel used per unit of energy produced by the reactor. Common metrics for evaluating *waste generation* include the mass, radioactivity, and/or volume of spent nuclear fuel, high-level waste, and/or low-level waste generated per unit of energy produced by the reactor.

on the technologies being investigated by DOE's Office of Nuclear Energy's (DOE-NE's) program, and other relevant programs, such as the Generation IV International Forum and the International Atomic Energy Agency. In its 2021 Strategic Vision, DOE-NE, the sponsor of this study, lists as two of its main goals "enable deployment of advanced nuclear reactors" and "develop advanced nuclear fuel cycles" (DOE-NE, 2021a).

The Nuclear Energy Innovation Capabilities Act of 2017 (NEICA) (Public Law 115-248) defines an *advanced nuclear reactor* as "a nuclear fission reactor with significant improvements over the most recent generation of nuclear fission reactors." NEICA states that these improvements may include "(1) inherent safety features, (2) lower waste yields, (3) greater fuel utilization, (4) superior reliability, (5) resistance to proliferation, (6) increased thermal efficiency, and (7) the ability to integrate into electric and nonelectric applications." This report addresses the potential merits of advanced nuclear reactors regarding waste, fuel utilization, and proliferation resistance, and focuses the safety assessment on the front and back ends of the fuel cycles, rather than on reactor operations. The other potential merits—reactor operational safety[2] and security, and the contribution of advanced nuclear energy to decarbonizing the U.S. energy system—will be addressed in a separate National Academies study, Laying the Foundation for New and Advanced Reactors in the United States, which is scheduled to conclude in early 2023.

In 2015, prior to the passage of NEICA, Congress directed DOE to conduct a planning study to evaluate "advanced reactor technology options, capabilities, and requirements within the context of national needs and public policy to support innovation in nuclear energy." The resulting report, *Advanced Demonstration and Test Reactor Options Study*, was published in 2017 (Petti et al., 2017). It defined advanced reactors as those that use nonwater coolants, "have the potential to expand the energy applications, enhance the competitiveness, and improve the sustainability of nuclear energy," and may include technology innovations such as

- "higher outlet temperatures than light water reactors (LWRs), which yield enhanced efficiency of electricity generation as well as for a variety of process heat applications;
- enhanced inherent safety, including passive decay heat removal systems;
- advanced fuels (liquid, particle, metallic, ceramic) and cladding enabling high burnup, extensive actinide destruction, and enhanced accident tolerance;
- advanced power conversion systems (Brayton cycle, supercritical CO_2) to improve overall energy conversion efficiency and reduce water usage;
- modular design to shorten construction times and to support phased deployment to allow flexibility in meeting demand; and
- greater degrees of autonomous control to minimize operating cost."

A team of stakeholders from industry, academia, and government carried out a range of technology readiness assessments to evaluate various advanced reactor concepts on the above criteria. While evaluations of reactor design and deployment criteria fall into the purview of the parallel National Academies study, it is important for this committee to understand DOE's thinking on promising reactor designs in order to analyze the associated fuel cycles.

In addition to the potential merits of advanced reactors defined in NEICA and Petti et al. (2017), DOE has established criteria and related objectives for its advanced fuel cycle and waste management programs, most notably in the 2014 *Nuclear Fuel Cycle Evaluation and Screening—Final Report* (*NFCE&S*) (Wigeland et al., 2014) and the Advanced Research Project Agency-Energy's Optimizing Nuclear Waste and Advanced Reactor Disposal Systems (ONWARDS) program. With a goal of informing DOE-NE's research and development priorities, *NFCE&S* evaluated fuel cycle options against the current LWR once-through cycle based on nine criteria (Wigeland et al., 2014). Six of these criteria are related to potential benefits that could be achieved in implementing an advanced fuel cycle: nuclear waste management, proliferation risk, nuclear material security risk, safety, environmental impact, and resource utilization. The other three criteria focus on challenges associated with deploying a new fuel cycle: development and deployment risk, institutional issues, and financial risk and economics.

The ONWARDS program aims to demonstrate technologies for the back end of the fuel cycle that reduce waste disposal impacts, with an ultimate goal of a 10-fold reduction in waste volume or repository footprint and disposal

[2] In some respects, reactor safety considerations are linked to fuel cycle parameters (such as fuel burnup limits).

SIDEBAR 1.1
Statement of Task

The National Academies of Sciences, Engineering, and Medicine will appoint an ad hoc committee of experts to evaluate and assess nuclear fuel cycles and technology options and the waste aspects of advanced nuclear reactors that could be commercially deployed by 2050. The committee will consider the relevant work performed by the fuel cycle program of the Department of Energy's Office of Nuclear Energy (DOE-NE), the various proposed advanced reactors investigated by DOE-NE, and other relevant programs. The committee will prepare a consensus report that will:

- Evaluate the merits and assess the viability of different nuclear fuel cycles, including fuel cycles that may use reprocessing, for both existing and advanced reactor technology options by:
  - Accounting for linkages among all elements of the fuel cycle, including waste transportation, storage, and disposal associated with the front and back ends of the fuel cycle, and concerns related to safety.
  - Examining the potential costs of the different nuclear fuel cycles required for advanced nuclear reactors.
- Evaluate nonproliferation implications and security risks of fuel cycles for advanced reactors by:
  - Including assessments of high-assay low-enriched uranium, uranium-plutonium mixed oxide fuel, and advanced fuel cycles that require separating plutonium from spent fuel.
  - Examining nuclear material accounting and control as well as containment, surveillance, monitoring, and timeliness of detection of diversion.
  - Accounting for how these can be addressed by International Atomic Energy Agency safeguard activities.
- Evaluate the waste management and disposal options for the various proposed advanced nuclear reactors by:
  - Accounting for typical volumes and physical, chemical, and isotopic characteristics of waste streams, including from possible reprocessing, from these advanced nuclear reactor technologies.
  - Examining transportation, storage, and ultimate disposal requirements for these wastes.

The consensus report will provide findings and recommendations that may consider evidence-based policy options.

costs less than $1/MWh (megawatt hour) (Shafer, 2021). The program focuses on technology developments in fuel recycling processes, safeguards, and high-performance waste forms. In March 2022, as this report was being completed, the ONWARDS program had awarded $36 million to 11 programs for such technology development, specifically targeting waste management associated with advanced reactors and fuel cycles (DOE, 2022a).

The committee received briefings on both *NFCE&S* and the ONWARDS program and considered their criteria and objectives in its analysis. The remainder of this chapter describes the status of U.S. nuclear reactors and fuel cycles to serve as a baseline against which advanced reactor technologies and associated fuel cycle and waste management strategies can be compared, summarizes previous National Academies' reports relevant for this study's topics, and provides a roadmap for the structure of the report.

1.2 NUCLEAR FUEL CYCLE DEFINITIONS AND THE BASIS SET OF FUEL CYCLES FOR THIS REPORT

The nuclear fuel cycle involves the passage of nuclear fuel through a series of stages. The front end involves the preparation of the fuel, leading to the service period, in which the fuel is used during reactor operations. The back end involves activities necessary to safely manage, contain, and either reprocess and recycle the fissile iso-

topes remaining in the fuel or directly dispose of spent nuclear fuel. In a 2021 study, the Nuclear Energy Agency of the Organisation for Economic Co-operation and Development (NEA-OECD) (2021) explored the "nuclear fuel cycle options—combination of nuclear fuel types, reactor types, spent nuclear fuels treatments and disposal schemes" being contemplated by countries with active nuclear power programs. The study looked at differing characteristics for both the development and implementation of a particular option, as well as the decision drivers countries face, such as the size of their nuclear program, including research and development (R&D), extended storage and disposal of waste, safety and environmental risks, needed supporting infrastructure, and cost—to name a few. Because of the vast number of options that exist, the NEA-OECD group found it convenient and easier to understand to reduce the fuel cycles to a basis set of only three:

- Open cycle systems (also known as the once-through fuel cycle) use low-enriched uranium in LWRs and dispose of the spent nuclear fuel directly in a deep geologic repository (see Figure 1.1).
- Monorecycle systems involve a single cycle of reprocessing spent nuclear fuel to (1) separate plutonium, which is combined with depleted uranium to produce mixed oxide fuel for use in LWRs; and (2) reprocess uranium, which can be reenriched and used in LWRs, although this is rarely done because a dedicated enrichment cascade is required, as two uranium isotopes are more radioactive and increase the dose rate. High-level radioactive waste from reprocessing (e.g., fission products and minor actinides) and the spent mixed oxide and reprocessed uranium fuels are disposed of in a deep geologic repository.
- Multirecycle systems involve reprocessing spent fuel repeatedly and recycling separated plutonium (and possibly minor actinides, such as neptunium) in fresh fuel. In principle, only high-level radioactive waste from reprocessing (composed of fission products, as well as actinides that are not separated for use in fuel) would need to be disposed of in a deep geologic repository as long as the fuel cycle is active (NEA-OECD, 2021).

This basis set—open cycle, monorecycle, and multirecycle systems—is used throughout the remainder of this report to simplify the discussion of fuel cycles, refined as necessary to address specific technologies. For example, many advanced reactor developers are planning initial implementation of an open fuel cycle for their designs, which use different fuel types and enrichments than the current LWR fleet. LWRs in operation today use either an open cycle or monorecycle system, as further discussed in Chapters 2 and 4. The following section of this chapter summarizes the status of the U.S. nuclear power program that relies on LWRs and the once-through fuel cycle. The multirecycle fuel cycle, which would require introducing advanced reactors into the mix of reactor technologies, is explored in Chapters 3 and 4.

1.3 STATUS OF THE U.S. NUCLEAR POWER PROGRAM

The United States has deployed commercial nuclear power since the 1950s, and as of 2021, nuclear power accounts for approximately 20 percent of U.S. electricity generation. The current commercial nuclear fleet consists entirely of thermal LWRs operating with low-enriched uranium dioxide fuel in a once-through fuel cycle. The LWRs operate on uranium dioxide fuel enriched to 3–5 percent uranium-235 and use light water as the coolant and moderator.[3] LWRs typically achieve power outputs of up to 3,800 MWth (megawatts thermal) or up to 1,250 MWe (megawatts electric),[4] corresponding to a thermal efficiency of 33 percent (DOE, 2015a; Murray and Holbert, 2020). There are two main types of LWRs: pressurized water reactors and boiling water reactors, which differ in method and location of steam production (U.S. NRC, 2020b). In a pressurized water reactor, the heat from the primary coolant loop is transferred to a secondary loop via heat exchangers (referred to as steam generators), in

[3] The moderator is a low–atomic number material that interacts by colliding with fission neutrons to reduce their energy to more efficiently induce the fission reaction with certain fissile isotopes. For example, uranium-235 fissions more easily with thermal neutrons (neutrons with around 0.025 eV [electron volts] of energy).

[4] Several LWRs achieve higher power outputs; for example, Grand Gulf Nuclear Station in Mississippi has a power output over 1,400 MWe; the Barakah Power Plants in the United Arab Emirates each have a power output of 1,400 MWe; and EPR, with operational plants in China and Finland and planned reactors in France and the United Kingdom, has a power output of 1,600 MWe.

which the water in the secondary loop is transformed into high-pressure steam that drives the electricity generator via a turbine. In contrast, boiling water reactors have only one coolant loop, in which the water coolant absorbs heat from the reactor core, causing it to boil and yielding a steam–water mixture. This mixture then undergoes a separation process to generate pure steam that powers the turbine/generator set to provide electricity. In the United States, about two-thirds of the operating reactors are pressurized water reactors and one-third are boiling water reactors.

All operational nuclear reactors are supported by a fuel cycle, which includes the activities required to fuel the reactor and manage its spent nuclear fuel upon discharge from the reactor. Figure 1.1 depicts the open, or once-through, fuel cycle employed in the United States, with a dashed box around the geologic repository to indicate that none yet exist for spent fuel disposal. For use in an LWR, natural uranium, which contains about 0.7 percent of the only naturally occurring fissile nuclide, uranium-235, has to be enriched to 3–5 percent uranium-235 to provide sufficient fissile material to achieve criticality.[5] Burnup, another important concept, measures the amount of energy produced by a unit of uranium mass, typically expressed as gigawatt-days per metric ton (GWd/MT) of uranium. Greater enrichment levels can provide for greater burnups.

The first steps to making enriched uranium fuel involve conversion of mined and milled uranium ore (U_3O_8) to uranium hexafluoride (UF_6) gas, a compound suitable for the enrichment process in gas centrifuges.[6] UF_6 can readily undergo phase changes at temperatures and pressures compatible with relevant industrial processes. It is used in gaseous form during enrichment, in liquid phase for transfer between containers and equipment, and as a solid for storage. The enriched uranium is converted back to an oxide (UO_2) and then fabricated into assemblies of fuel rods that make up the reactor core. (Because of the chemical stability of UO_2, it is desirable to eventually deconvert UF_6 back to UO_2 prior to disposal of the enrichment tails.) Irradiation of the fuel in the reactor for 4–6 years depletes the fissile uranium in the assemblies to a point at which the fuel is no longer suitable for power production; the spent fuel is transferred into a water-filled pool (i.e., the spent fuel pool), where it will be stored for several years.

The spent uranium oxide (UOX) nuclear fuel[7] discharged from a reactor mostly contains actinides and fission products. See Sidebar 1.2 for information about some of these radionuclides and their relative masses prior to and following irradiation of 1 MT of uranium (as UOX) enriched at 3.3 percent to a burnup of 33 GWd/MT. A burnup of 33 GWd/MT is typical of much of the spent fuel in the U.S. inventory; however, more recent spent fuel has a burnup of over 45 GWd/MT because of the practice of moving toward higher burnup, as well as enrichment of up to 4.8 percent (WNA, 2021c). Higher burnup allows utilities to extract more power from the fuel before replacing it, which translates to longer operating periods between refueling and the use of fewer fuel assemblies (U.S. NRC,

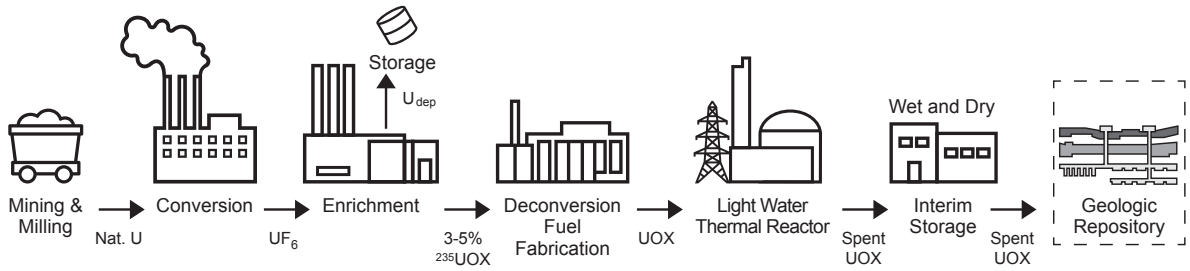


FIGURE 1.1 Schematic of an open or once-through fuel cycle for a light water reactor (LWR).
NOTE: Nat. U = natural uranium; U_{dep} = depleted uranium; UF_6 = uranium hexafluoride; UOX = uranium oxide.
SOURCE: Adapted from MIT (2011).

[5] Note that some accident-tolerant fuel designs being considered for use in LWRs have uranium-235 enrichments of up to 10 percent (U.S. NRC, 2020a).

[6] Gaseous diffusion using UF_6 was the first technology used commercially for uranium enrichment. As the more efficient gas centrifuge technology became available, it replaced gaseous diffusion.

[7] This report uses the term *spent nuclear fuel* for all fuel discharged from a nuclear reactor. Some reports (e.g., NEA-OECD, 2021) differentiate between *spent fuel*, if the fuel will be disposed as waste, and *used fuel*, if the fuel will be recycled.

SIDEBAR 1.2
Radionuclides in Spent Nuclear Fuel

In addition to the spent uranium oxide (UOX) discharged from a reactor, spent fuel includes cladding material and the fuel assembly hardware. The spent UOX itself mostly contains actinides and fission products. The major actinides are uranium and plutonium, and the minor actinides include neptunium, americium, and curium. Fission products are elements formed from the fission of heavy nuclei (e.g., ^{235}U, ^{239}Pu). All actinides and most fission products are radioactive; the rate of heat generation from radioactive decay decreases over time as the quantity of radioactive material decays. As depicted in the figure below, spent UOX nuclear fuel discharged from a 1,000-MWe pressurized water reactor after 3 years of operation contains approximately 95 percent ^{238}U, 1 percent ^{235}U, 0.9 percent Pu, 0.1 percent minor actinides, and 3 percent fission products. The most abundant radionuclides in spent nuclear fuel, along with their half-lives, are shown in the table below. The exact radionuclide content of spent nuclear fuel depends on the fuel type and reactor operating conditions. The burnup, or total energy extracted per unit mass of fissionable fuel, is an important factor for decay heat and in producing long-lived nuclides.

Radionuclides can be classified as (1) short-lived radionuclides, with a decay half-life equal to or shorter than 30 years, and (2) long-lived radionuclides, with a decay half-life longer than 30 years. Most fission products are short lived (CEA, 2008). After several years of cooling, the radioactivity and decay heat from spent nuclear fuel is dominated by the decay of cesium-137 and strontium-90, which are produced in about 6 percent of fissions and have half-lives of about 30 years. In waste containing only short-lived radionuclides, the radioactivity originally present will be reduced to less than 0.1 percent of its initial value after 10 half-lives, or 300 years. Given the same number of atoms, long-lived radionuclides emit radiation at a lower rate than short-lived radionuclides and are characterized as having lower activity (i.e., amount of ionizing radiation released by a material as a function of time, or decay per unit time). Long-lived nuclides include a few long-lived fission products and some isotopes of uranium, plutonium, and minor actinides, as shown in the table below.

TABLE Most-Abundant Isotopes of Uranium (U) and Plutonium (Pu), Minor Actinides, and Long-Lived Fission Products Found in Spent UO_2 Light Water Reactor Fuel, and Their Half-Lives[a]

U and Pu	Half-Life (y)	Minor Actinides	Half-Life (y)	Long-Lived Fission Products	Half-Life (y)
^{235}U	7.04×10^{8}	^{237}Np	2.14×10^{7}	^{79}Se	3.27×10^{5}
^{236}U	2.34×10^{7}	^{241}Am	4.33×10^{2}	^{93}Zr	1.53×10^{6}
^{238}U	4.47×10^{9}	^{243}Am	7.36×10^{3}	^{99}Tc	2.11×10^{5}
^{238}Pu	8.77×10^{1}	^{243}Cm	2.9×10^{1}	^{107}Pd	6.5×10^{6}
^{239}Pu	2.41×10^{4}	^{244}Cm	1.81×10^{1}	^{126}Sn	2.3×10^{5}
^{240}Pu	6.56×10^{3}	^{245}Cm	8.42×10^{3}	^{129}I	1.57×10^{7}
^{241}Pu	1.42×10^{1}	^{246}Cm	4.71×10^{3}	^{135}Cs	2.3×10^{6}

[a] The most abundant isotopes for other fuels, such as fissile Pu in depleted U, fissile ^{233}U in natural thorium, and fissile Pu in natural thorium vary (Croff and Krahn, 2016).

SOURCES: Generated by the committee using data from IAEA (2022) and IN2P3 (n.d.-b).

2018a). As a result of its radioactive decay, spent fuel releases heat and is stored in spent fuel pools after discharge from the reactor to help dissipate this heat.

Initially, cooling pools at reactors were sized with the expectation that the spent fuel would be reprocessed. Since reprocessing never became a realistic option in the United States and a repository is not yet available, reactor operators have been faced with the need for additional storage capacity for spent fuel. Initial pool capacity expansion was achieved with the use of neutron flux traps and by reracking operations, but there are limits to these approaches. Dry cask storage technology was developed in the 1980s to cope with the pool capacity limitations, allowing spent

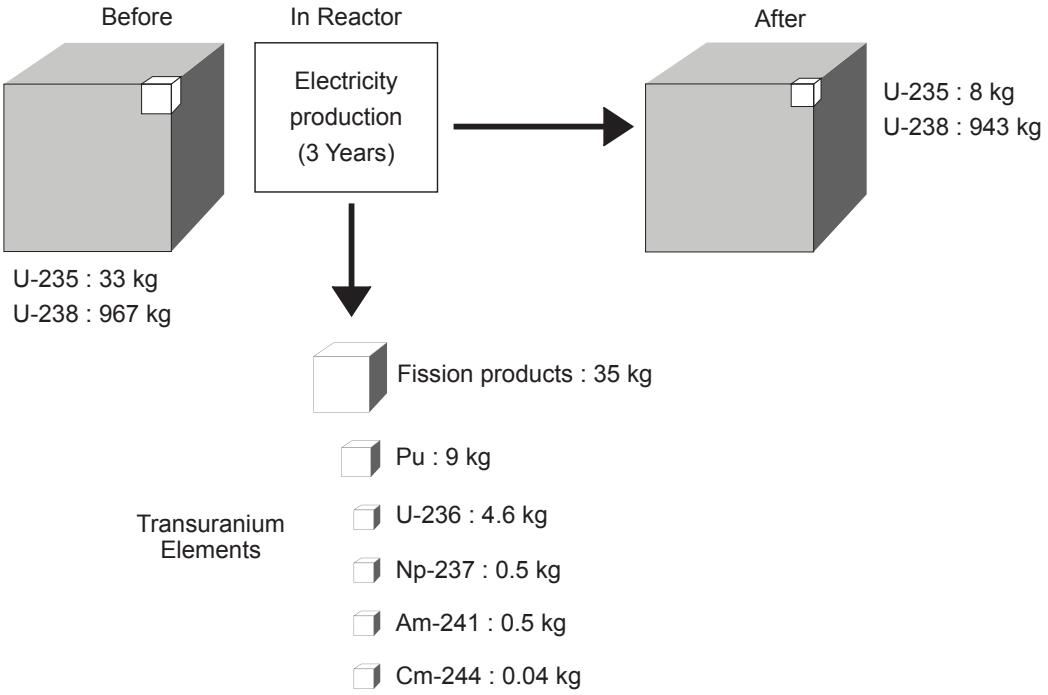


FIGURE Typical composition of spent nuclear uranium oxide fuel from a 1,000-MWe pressurized water reactor with an initial charge of 1000 kg of U with ^{235}U enriched to 3.3 percent after roughly 3 years of operation (burnup 33 GWd/MT).
NOTE: Not depicted in the figure are relatively small amounts of activation products, such as ^{14}C, ^{36}Cl, ^{59}Ni, and ^{60}Co, that form when neutrons originating in the fission process are absorbed by structural materials in the reactors, such as metals and their associated impurities in the fuel assembly.
SOURCE: Adapted from IN2P3 (n.d.-a). Courtesy of Isabelle Billard.

fuel to be dried and transferred to specially designed casks after about 5 years of water cooling. Dry storage casks manage decay heat from the spent fuel through a combination of conduction, convection, and radiation to maintain the internal environment of the cask within design specifications. Dry cask storage has the further advantage of freeing up space in a cooling pool to accommodate freshly discharged spent fuel from the reactor. At present, most wet and dry cask spent fuel storage is carried out at the reactor site pending eventual long-term disposal, likely in a deep-mined geologic repository (BRC, 2012; U.S. NRC, 2017). As there is currently no long-term disposal available for spent fuel in the United States, more than 86,000 MT of spent fuel are currently in wet or dry storage at 75 operating or shut-down reactor sites in 33 states; this figure grows by 2,000 MT per year (GAO, 2021a).

1.3.1 Status of the Front End of the Fuel Cycle in the United States

The front end of the fuel cycle includes the mining or extraction of uranium ore, milling of the uranium ore to produce U_3O_8, chemical conversion to UF_6, enrichment to increase the content of uranium-235, fabrication into fuel assemblies of UO_2, and delivery of the assemblies to the reactor (U.S. NRC, 2017). A 1,000-MWe LWR operating at 100 percent capacity requires approximately 200 MT of natural uranium per year. The majority of U.S. uranium is imported to the United States from Canada and Australia (Larson, 2019; MIT, 2011). Notably, imports from Russia are limited strictly and currently represent about 16 percent of U.S. demand. In 2019, domestic production of uranium concentrate totaled 0.17 million pounds (approximately 80 MT), a nearly 90 percent decrease from the previous year (EIA, 2020). No conversion facilities are currently operating in the United States; the primary sources of UF_6 imports are Canada and the United Kingdom (Larson, 2019). However, the Honeywell conversion facility in Illinois, which halted operation in 2018, announced plans to come back online in 2023 (Patel, 2021). The sole U.S.-based enrichment facility, which is foreign owned and operated by Urenco USA, provides about one-third of the enriched UF_6 required for U.S. reactors, with the remainder being imported from the Netherlands, Germany, Russia, and the United Kingdom (Larson, 2019). A commercial LWR requires about 20 MT of fuel per year, and three facilities capable of fabricating LWR uranium oxide fuel currently operate in the United States to meet the fueling needs of commercial U.S. reactors: Global Nuclear Fuel-Americas in Wilmington, North Carolina; Westinghouse Columbia Fuel Fabrication Facility in Columbia, South Carolina; and Framatome, Inc., in Richland, Washington (MIT, 2011; U.S. NRC, 2020c).

1.3.2 Status of the Back End of the Fuel Cycle in the United States

The back end of the fuel cycle encompasses the management of spent fuel, including interim storage, waste transportation, final geologic disposal, and—in certain countries (see Chapter 2 and Appendix H for more information)—reprocessing (chemical separation techniques) to separate fissile materials from other spent fuel constituents. In the initial stages of its commercial nuclear power development, the United States intended to establish a fuel cycle based on recovering the plutonium present in the LWR spent fuel, using reprocessing, and fueling a planned fleet of fast neutron spectrum (or simply "fast") reactors. The U.S. intent, shared by many other international programs, originated from the perception that natural uranium resources, especially the supply of uranium-235, were scarce compared with the projected large number of LWRs that would be required to meet anticipated global electricity demand. However, as will be further discussed in the next chapter, several factors—including technical challenges, nonproliferation violations, nuclear accidents, changes in electricity market conditions, unfavorable economics of reprocessing, and expanded natural uranium resources—have resulted in abandonment of the plutonium-fueled fast reactor technology option in the United States and many other countries and have limited progress in the few countries that have continued to pursue such a program. Because the United States does not currently have any centralized interim spent fuel storage facilities or reprocessing plants, nor a deep geologic repository for final disposal, spent nuclear fuel remains at nuclear power plant sites, making at-reactor long-term dry storage the de facto endpoint of the current U.S. fuel cycle.

The Nuclear Waste Policy Act (NWPA) (Public Law 97-425) set the ultimate strategy for commercial spent fuel as disposal in a deep geologic repository, and the 1987 Amendments to that act (Public Law 100-203, Part E) selected Yucca Mountain, Nevada, as the only site to be examined for disposal. The U.S. Nuclear Regulatory Commission (U.S. NRC) developed regulations for waste disposal at Yucca Mountain based on radiation dose standards set by the U.S. Environmental Protection Agency (EPA). Under the NWPA, DOE committed to starting the transport of spent fuel from storage at nuclear power plants to a geologic disposal site by 1998, but it has missed that deadline by nearly 25 years so far. In 2008, DOE submitted a license application to the U.S. NRC to construct the Yucca Mountain repository. Although DOE filed a motion in 2010 to withdraw the application, it was denied by the Licensing Board, which, supported by the District of Columbia Court of Appeals in 2013, instructed the U.S. NRC to complete the license review. In 2015, U.S. NRC staff completed the Safety Evaluation Report, which found that the license application met regulatory requirements, pending additional licensing steps. Nonetheless, DOE terminated its technical support for the statutorily required Office of Civilian Radioactive

Waste Management, which was in charge of managing spent fuel and siting a geologic repository. Congress has not appropriated any funds to move forward on licensing for Yucca Mountain since 2010.

The 2012 Blue Ribbon Commission on America's Nuclear Future (BRC) identified several factors contributing to the delays and challenges in establishing a repository at Yucca Mountain. These included a perceived lack of technical and scientific considerations behind site selection, opposition from the state of Nevada and its citizens, unrealistic deadlines set by DOE, and inflexibility in the process and guidelines established by the NWPA Amendments (BRC, 2012). For future attempts to site a nuclear waste disposal facility, at Yucca Mountain or elsewhere, the BRC recommended a consent-based, transparent, adaptable approach involving phased decision making and establishing a safety case for the site (BRC, 2012). In 2018, the Steering Committee on the Reset of America's Nuclear Waste Management Strategies and Policies similarly recommended a consent-based siting approach for establishing a geologic repository, emphasizing the need to collaborate with local communities and to allow for flexibility throughout the process (Reset Steering Committee, 2018). Both the BRC and Reset reports recommended the establishment of a new, sole-purpose, independent waste management and disposal organization but differed on the form of such an organization. In Chapter 5, the committee further examines this concept and develops its related findings and recommendations for nuclear waste management and disposal. As of the writing of this report, political opposition to Yucca Mountain remains, and there are no actionable plans to site a repository in the United States.

1.4 RELATED NATIONAL ACADEMIES STUDIES

The National Academies have published a number of reports on issues related to this study, including advanced nuclear energy (fission and fusion), spent fuel and high-level waste treatment, nuclear waste management, and nonproliferation. The committee considered the results of these prior reports in its analysis. Brief summaries of select reports published in the past 30 years, including major findings, recommendations, and conclusions related to this study, are presented below.[8] The National Academies have a long history of providing advice to the federal government on nuclear waste management and disposal, dating back to the seminal 1957 report *The Disposal of Radioactive Waste on Land*, which identified deep geologic disposal as the most promising option for safe disposal of radioactive wastes. The collection of summaries below includes only several of the numerous National Academies studies on the topic of nuclear waste management; a more complete list can be found in Appendix F.

As mentioned above, in parallel to this study, a separate National Academies committee is evaluating "opportunities and barriers to the commercialization of new and advanced nuclear reactor technologies in the United States over the next 30 years as part of a decarbonization strategy." That study, with an anticipated conclusion in early 2023, focuses on the technical viability of advanced reactor technologies, including their potential use for nonelectricity applications, as well as the economic, regulatory, market, and societal challenges for commercialization. The full statement of task for that study is reproduced in Appendix I.

Bringing Fusion to the U.S. Grid (2021)

This report identifies the key goals and innovations needed to develop a fusion pilot plant in the United States. Considering input from electric utilities, the committee recommended that, for fusion to play a role in the transition to a low-carbon electricity system, net electricity production from a fusion pilot plant should occur in 2035–2040. Meeting that aggressive timeline, the committee concluded, would require urgent investments by both DOE and the private sector, possibly including public–private partnerships. To that end, the committee recommended the creation of national teams with representatives from industry, academia, and national laboratories to develop conceptual pilot plant designs and technology roadmaps.

[8] All of these National Academies reports are available for free at https://nap.nationalacademies.org.

Improving the Assessment of the Proliferation Risk of Nuclear Fuel Cycles (2013)

This study analyzes the use of technical assessments of proliferation risk to (1) inform R&D decisions about nuclear fuel cycles and nonproliferation policy and (2) improve communication of those decisions to stakeholders and the public. Because predefined frameworks had previously been poorly implemented and do not address extrinsic factors that can change over time, the committee recommended considering a probabilistic risk assessment approach as an alternative. On proliferation considerations for future fuel cycle decisions, the committee recommended "that fuel cycle R&D decisions include proliferation resistance (rather than proliferation risk) as one factor among others (such as cost and safety) to guide those decisions" and "that DOE-NE and NNSA [National Nuclear Security Administration] jointly decide upon a set of high-level questions comparing the proliferation resistance of proposed future fuel cycles to the current once-through fuel cycles to determine as early as possible in their development whether the former have significantly different intrinsic proliferation resistance (either for the better or for the worse)."

America's Energy Future: Technology and Transformation (2009)

This study evaluates current (2009) and projected technologies for energy supply, storage, and end use, with particular consideration of times to readiness for deployment; R&D challenges; estimated costs and performance; and impacts on environmental, economic, policy, social, and national security factors. Nuclear energy is covered in Chapter 8 of that publication. Regarding advanced fuel cycles, the report finds

> Considerable R&D is needed before alternative fuel cycles will be ready for deployment. It is prudent to pursue such R&D, which is likely to be resource intensive and time-consuming, but to not initiate facility construction at present. Increasing proliferation resistance as well as reducing the cost of fuel cycle processes and associated facilities will be a major goal of the R&D effort. Commercial-scale facilities are unlikely to be ready for deployment until after 2035.

Internationalization of the Nuclear Fuel Cycle: Goals, Strategies, and Challenges (2009)

Produced by a joint committee of the U.S. and Russian Academies of Sciences, this study analyzes the possible internationalization of the nuclear fuel cycle to meet nonproliferation and fuel assurance goals. Finding that uranium enrichment and spent fuel reprocessing are the primary concerns for producing direct-use materials, the report recommends that (1) countries currently providing nuclear fuel should ensure a stable supply to disincentivize other nations from developing enrichment capabilities and (2) the international community should provide adequate storage capacity and/or reprocessing services for spent fuel to limit the spread of reprocessing technology. Furthermore, the committee recommends that development of new reprocessing technologies should occur in parallel with assessments of the technologies' costs and proliferation risks, that R&D should be performed on advanced safeguards and security technologies, and that "spent fuel should only be reprocessed when its constituents are needed for fuel, or when reprocessing is necessary for safety reasons."

Review of DOE's Nuclear Energy Research and Development Program (2008)

This study evaluates program goals and plans for DOE-NE and recommends policies and research activities to "advance NE's mission of securing nuclear energy as a viable, long-term commercial energy option to provide diversity in energy supply." The report focuses on six activities: Nuclear Power 2010, the Generation IV Program, the Nuclear Hydrogen Initiative (NHI), the Advanced Fuel Cycle Initiative (AFCI), the Global Nuclear Energy Partnership (GNEP) program, and DOE-NE's collaboration with Idaho National Laboratory (INL). The committee made recommendations related to each program and prioritized the activities based on its judgment of how each supports DOE-NE's overall mission. High priority was given to the Nuclear Power 2010 program and university infrastructure support; medium priority was given to Generation IV, NHI, AFCI, and INL programs; low priority was given to facility deployment in GNEP. The committee further recommended,

> As a counterbalance to the short-term nature of the federal budget process, NE should adopt an oversight process for evaluating the adequacy of program plans, evaluating progress against these plans, and adjusting resource allocations as planned decision points are reached.

It should be noted that two committee members provided a dissenting opinion on the recommendations for pursuing a reprocessing and fast reactor R&D program under ACFI and for DOE having a role in commercializing fuel cycle technologies.

Going the Distance?: The Safe Transport of Spent Nuclear Fuel and High-Level Radioactive Waste in the United States (2006)

This study analyzes technical and societal concerns for the transportation of spent nuclear fuel and high-level radioactive waste (SNF/HLW) in the United States, in the context of federal plans to develop a permanent geologic repository at Yucca Mountain and a commercial interim storage facility in Utah. The committee found that, while there are no fundamental technical barriers to safe transport, there may be social and institutional barriers, and the social risks present a challenge to the implementation of an SNF/HLW transportation program. Transportation packages generally provide a robust barrier to radionuclide release, but extreme accidents like long-duration fires could compromise their effectiveness. Except in these rare extreme accident scenarios, the radiological health and safety risks of SNF/HLW transportation are low. Based on its findings, the committee recommended that

- U.S. NRC perform additional analysis of long-duration fire scenarios and implement controls and restrictions on shipments based on the results;
- full-scale testing of transportation packages continue;
- transportation planners (1) identify and mitigate potential hazards along transportation routes that could lead to extreme accidents and (2) establish mechanisms to gather information and advice about social risks;
- the U.S. Department of Transportation ensure that designated shipment routes comply with regulatory requirements;
- DOE fully implement its decision to ship SNF/HLW to the repository by rail on dedicated trains and identify and publicize its preferred transportation routes;
- DOE and commercial spent nuclear fuel owners negotiate to ship older fuel first to a repository or interim storage facility; and
- DOE and Congress examine organizational changes to DOE's SNF/HLW transportation program and consider as options (1) a quasi-independent office within DOE, (2) a quasi-governmental corporation, or (3) a fully private organization.

The committee also notes that, given its lack of access to classified information, it did not examine malevolent acts against SNF/HLW shipments; however, it recommended an independent analysis of these transportation security issues.

Monitoring Nuclear Weapons and Nuclear-Explosive Materials: An Assessment of Methods and Capabilities (2005)

This report analyzes approaches in transparency and monitoring that can be used to verify nuclear weapons and nuclear-explosive materials. The report defines *nuclear-explosive materials* as (1) a mixture containing uranium-235 and uranium-238 with more than 20 percent uranium-235, (2) a mixture of uranium-233 and uranium-238 with more than 12 percent uranium-233, or (3) a mixture of plutonium isotopes with less than 80 percent plutonium-238; but it also notes that "nuclear explosives can in principle be made with material containing somewhat less than 20 percent U-235, but the amount of material required at enrichments below 20 percent is very large." The study concludes that technical tools are available to improve transparency and monitoring measures throughout the nuclear weapon life cycle and for nuclear-explosive materials and that they can be implemented

under existing international agreements to decrease uncertainties in assessing foreign stockpiles of nuclear weapons and materials.

One Step at a Time: The Staged Development of Geologic Repositories for High-Level Radioactive Waste (2003)

This report advises DOE on staged implementation of repository development throughout the construction, operation, closure, and postclosure phases and examines the associated programmatic, safety, security, institutional, regulatory, and societal issues. The study analyzes two approaches to staging: *linear staging*, in which there is "a single, predetermined path to a selected, well-defined endpoint, where stages are defined principally as milestones at which costs and schedules are reviewed and modified as necessary," and *adaptive staging*, which provides a reference framework for the project but maintains flexibility using a deliberate decision-making process between stages "to guide the implementer in identifying program improvements with respect to, for instance, safety, environmental impacts, costs, and schedules." The committee recommended that the adaptive staging approach be used, both for a generic repository program and specifically for the Yucca Mountain case, believing it to be more effective and less error-prone than linear staging. The committee further recommended that any repository program should (1) reevaluate safety at each decision point in a structured decision-making process; (2) use learning opportunities from in situ testing (e.g., in pilot, test, or demonstration facilities); (3) continually and actively incorporate learning during the repository operation period; and (4) incorporate independent technical advice and stakeholder input via a technical oversight group and stakeholder advisory board, respectively.

Disposition of High-Level Waste and Spent Nuclear Fuel: The Continuing Societal and Technical Challenges (2001)

This report addresses the questions of *whether* and *when* to implement geologic disposal of high-level waste, aiming to help policy makers with decision making and inform the interested public. The committee found that monitored surface storage and geologic disposition are feasible options to handle the inventory of high-level waste, with the latter being the only long-term solution. Furthermore, the greatest barrier to waste disposition is lack of public support; individual countries have to decide whether, when, and how to move forward with a repository; and successful approaches involve stepwise processes, transparent and participatory decision making, and international cooperation. The committee recommended the formation of national organizations responsible for management of high-level waste that address technical and safety issues with project development; involve the public in decision making; develop stepwise programs with realistic alternative options; engage in international cooperation on standards and strategies; and perform integrated, comprehensive, and risk-based analyses to ensure the safety and security of waste management facilities.

Electrometallurgical Techniques for DOE Spent Fuel Treatment: Final Report (2000)

This study evaluates the technical viability of electrometallurgical processing of spent nuclear fuel, focusing on the treatment of sodium-bonded spent fuel from the Experimental Breeder Reactor-II (EBR-II). This 10th and final report in a series of studies on the topic analyzes waste streams and waste form options for electrometallurgically treated EBR-II fuel; evaluates the EBR-II Spent Nuclear Fuel Treatment Demonstration Project on criteria related to process, waste streams, and safety; and recommends postdemonstration activities if the same treatment is used for the remaining spent fuel. The committee noted no technical barriers to electrometallurgical treatment of EBR-II spent fuel, but emphasized that waste form qualification would be required before applying the process to other spent fuel inventories. The committee recommended, "If the DOE decides to treat the remaining sodium-bonded spent fuel inventory and the waste form qualification efforts are successful, the required equipment upgrades and facility modifications should be adequately funded to ensure that treatment can be completed in a reasonable time and at a reasonable cost."

Nuclear Wastes: Technologies for Separations and Transmutation (1996)

This report evaluates separation and transmutation systems as alternatives to the once-through fuel cycle and reviews options for processing high-level waste from defense programs. It considers the technical feasibility, ability for system integration, economics and financing methods, effect on repository capacity, proliferation risk, public acceptance, health impacts, and R&D needs of three transmutation concepts: (1) an LWR with a thermal neutron spectrum to transmute transuranic elements and some fission products; (2) an advanced liquid metal reactor with a fast neutron spectrum as a transuranic burner; and (3) an accelerator-driver subcritical nuclear reactor to reduce transuranic waste to Class C or lower levels. The committee reached four primary conclusions:

- Implementing any of these advanced separation and transmutation systems would not eliminate the need for a geologic repository for spent LWR fuel.
- The once-through LWR fuel cycle for commercial reactors should be continued.
- The time period for fuel retrievability should be extended to facilitate future implementation of alternative fuel cycle strategies.
- The United States should perform "a sustained but modest research and development program" on separation and transmutation technologies for spent fuel and defense waste to improve their cost effectiveness for potential future deployment.

Technical Bases for Yucca Mountain Standards (1995)

Mandated in Section 801(a)(2) of the Energy Policy Act of 1992, this study provides advice to EPA on the scientific basis for standards for deep geologic disposal of high-level radioactive waste at Yucca Mountain. The committee recommended the following:

- EPA should use an individual risk-based standard, rather than a dose-based standard, to limit negative health impacts from radiation release.
- A critical-group approach should be used to set the Yucca Mountain standards, where the group represents a relatively homogeneous set of individuals who would face the highest risk upon radiation release.
- The compliance assessment should be performed for the time of peak risk within the timeline of geologic stability of the repository, or around 1,000,000 years.
- EPA should require the estimated risk for a future intrusion scenario be no greater than the risk limit used for an undisturbed repository.

The committee also notes that the study only addressed the scientific basis for a repository standard and not the social, economic, and political aspects that must also be considered.

Nuclear Power: Technical and Institutional Options for the Future (1992)

Under the premise that nuclear power should continue to contribute to U.S. electricity supply, this report examines institutional and technological options for achieving that goal. The committee attributed the slowed growth of nuclear generation in the United States to reduced electricity demand; high cost; regulatory uncertainty; and public concerns around safety, economics, and waste disposal. It presented the following summary:

> The institutional challenges are clearly substantial. If they are to be met, the Committee believes that the Federal government must decide, as a matter of national policy, whether a strong and growing nuclear power program is vital to the economic, environmental, and strategic interests of the American people. Only with such a clearly stated policy, enunciated by the President and backed by the Congress through appropriate statutory changes and appropriations, will it be possible to effect the institutional changes necessary to return the flow of capital and human resources required to properly employ this technology.

The report also analyzes several advanced reactor technologies on criteria related to "safety in operation, economy of construction and operation, suitability for future markets, fuel cycle and environmental considerations, resistance to diversion and sabotage, technology risk and development schedule, and amenability to efficient and predictable licensing." The committee concluded that LWRs (both large evolutionary and midsized advanced designs) have the greatest potential to be cost effective. It recommended that Canadian Deuterium Uranium reactors, safe integral reactors, process inherent ultimate safety reactors, and modular high-temperature gas reactors should be low priorities for federal funding and that liquid metal reactors should receive high priority for long-term development despite their limited near-term market potential.

1.5 REPORT ROADMAP

To guide its response to the statement of task (see Sidebar 1.1), the committee developed several framing questions (see also Table 1.1 for a roadmap for the key topical areas covered in each chapter):

- What are the merits and viability of fuel cycle options for existing commercial light water reactors?
- What are the proposed merits of advanced reactors and fuel cycles? What are the relevant metrics to use in assessing those proposed merits to determine viability?
- What factors have the greatest impact on the viability of advanced reactors and fuel cycles?
- What factors affect having the requisite infrastructure for supplying the various advanced fuel types and for producing high-assay low-enriched uranium (HALEU), which almost all advanced reactor designs are projected to require?
- Do advanced reactors require reprocessing to achieve the proposed benefits related to waste reduction?
- Do advanced fuel cycles reduce the waste management problem, and if so, how?
- What will be the constituents of waste streams (including high- and low-level wastes, including Greater than Class C wastes) from advanced reactors, and how will these wastes impact storage, transportation, and disposal?
- What would be the costs of developing the required front- and back-end processes to support advanced reactors and fuel cycles?
- What are the safety considerations and risks for the front- and back-end processes required to support the development of advanced reactors and fuel cycles?
- What are the proliferation risks and security risks of advanced reactors and fuel cycles compared with the baseline once-through fuel cycle with LWRs?

These questions are addressed in Chapters 2–6 of the report, which is organized as follows:

Chapter 1 (this chapter) provides the background on the study request, sets the baseline for the current and projected status of the commercial U.S. nuclear power program, summarizes prior related work from the National Academies, and provides a roadmap for the report.

Chapter 2 considers the merits and viability of fuel cycle options for existing commercial reactor technologies. Beginning with an overview of the global development of nuclear power, including the initial rationale for development of fast breeder reactors, the chapter describes types of nuclear fuel cycles and examines various front- and back-end processes applicable to LWRs, with particular focus on the once-through and monorecycling fuel cycles.[9] The chapter also compares national policies related to nuclear fuel cycles, focusing on the experiences and lessons learned of France and the United States, the world's two leading nuclear power producers.

Chapter 3 describes the status and outlook for the advanced reactors under development. It provides background on the Generation IV International Forum's multinational work on six advanced reactor designs and details

[9] The committee highlights that the U.S. House of Representatives' Appropriations Committee initially requested a separate report on this topic but agreed at the first information-gathering meeting that the committee could include this task within this integrated report on advanced reactors and fuel cycle options.

U.S. efforts, focusing on DOE-NE's programs, to support the development of advanced reactors and associated fuel cycles, including the policy, economic, and regulatory factors for this development.

Chapter 4 examines the developments needed in the front and back ends of the fuel cycle to support potential deployment of advanced reactors and production of their associated fuel cycles and discusses safety and cost aspects of advanced fuel cycles.

Chapter 5 addresses the waste management and disposal options for three representative classes of advanced reactors: high-temperature gas-cooled reactors, sodium-cooled fast reactors, and molten salt reactors. In particular, it examines unique waste streams that would arise from these advanced reactors and their impact on storage, transportation, and geologic disposal.

Chapter 6 assesses the nonproliferation and security risks of fuel cycles associated with advanced nuclear reactors compared with the once-through cycle with LWRs.

Several key topical areas appear in multiple chapters of the report, as outlined in Table 1.1.

The appendixes provide brief biographies of the committee and staff (Appendix A), descriptions of the information-gathering meetings (Appendix B), a list of the acronyms and abbreviations (Appendix C), information on waste classifications (Appendix D), technical information on the flow sheets of representative advanced reactors and selected nuclear fuel cycle options (Appendix E), a list of National Academies reports on waste management (Appendix F), information on reprocessing and geologic disposal of TRIstructural ISOtropic (TRISO) fuel (Appendix G), information on reprocessing and recycling practices in other countries (Appendix H), and the statement of task for parallel National Academies' study Laying the Foundation for New and Advanced Nuclear Reactors in the United States (Appendix I).

TABLE 1.1 Roadmap for Discussions of Key Topical Areas for Advanced Reactors and Fuel Cycles

Topical Area	HALEU	TRISO Fuel Particles	Mixed Oxide Fuel	Once-Through Fuel Cycle	Monorecycling Fuel Cycle	Multirecycling Fuel Cycle	Reprocessing
Chapter 1				definition; status in the United States	definition	definition	
Chapter 2			use worldwide; experience in the United States	global status; U.S. policies	use in France; motivation and challenges for deploying this option		U.S. policies
Chapter 3	proposed use in advanced reactors	proposed use in advanced reactors		proposed use in advanced reactors		proposed use in advanced reactors	proposed use in advanced reactors
Chapter 4	enrichment, supply chain, cost, safety considerations	fabrication	fabrication		technical details	technical details, motivation and challenges for employing this option	options for different reactors and fuel cycles, technical details, cost estimates, safety considerations, challenges and potential benefits
Chapter 5	storage, transportation, and disposal issues	storage and transportation, considerations for geologic disposal		U.S. policy for waste management and disposal; associated waste streams for advanced reactors; storage and transportation			associated waste streams
Chapter 6	nonproliferation implications and security risks	nonproliferation implications and security risks	nonproliferation implications and security risks	nonproliferation implications and security risks	nonproliferation implications and security risks	nonproliferation implications and security risks	nonproliferation implications and security risks

NOTE: HALEU = high-assay low-enriched uranium; TRISO = TRistructural ISOtropic.

2

Merits and Viability of Existing Nuclear Fuel Cycles for U.S. Light Water Reactors

This chapter reviews the deployment and status of existing commercial nuclear reactors and associated fuel cycles with emphasis on the predominant light water reactors (LWRs) and the once-through and monorecycle fuel cycles. The chapter responds to the part of the first charge of the statement of task, which calls for an evaluation of the merits and assessment of the viability of different nuclear fuel cycles, including fuel cycles that may use reprocessing, for *existing* reactor technology options. The evaluation and assessment of fuel cycles for *advanced* reactor technology options are described in Chapters 3 and 4.

In this chapter, the committee provides the summary and findings up front (Section 2.1); describes the development of nuclear energy generation for civilian applications with emphasis on the United States, which pioneered the technology (Section 2.2); reviews the global status of the commercially established nuclear fuel cycle facilities (Section 2.3); and summarizes the not-yet-available facilities that would be needed to complete the existing full cycle options (Section 2.4). In Section 2.5, the committee uses a comparison of the situation in France—one of the two countries that currently operate commercial-scale reprocessing facilities—and in the United States—a country without such facilities—to illustrate the differing development of nuclear fuel cycle policies in these countries, which have the largest global share of nuclear power. Finally, in Section 2.6, the committee provides insights about the merits and viability of the fuel cycle options for LWR technologies.

2.1 CHAPTER 2 SUMMARY AND FINDINGS

At one time, the United States was a global leader in the development of LWR nuclear power technologies and in developing and testing some of the first non-LWR technologies. However, the lack of a long-term policy commitment and research and development (R&D) investment to nuclear energy has resulted in a partial decline in U.S. technical leadership and expertise. In contrast, China, France, and Russia, among other countries, have set clear national goals with regard to the role of nuclear energy in an overall national energy-security context and have supported the industry and nuclear R&D community for decades, in order to develop advanced LWR nuclear power technologies. These multidecade investments have led to the development of more technically complex technologies; even then, it is still unclear that these technologies will be available by 2050. Some limited progress is apparent, such as Russia's BN-600 and BN-800 fast reactors and France's La Hague reprocessing facility for LWR spent fuel, but there are many more instances where development of alternatives to the LWR-based once-through fuel cycle has been halted or delayed for decades for various reasons, including cost proliferation and concerns.

Monorecycling of plutonium in LWRs in France and other countries was born out of the need to mitigate or minimize the buildup in plutonium inventory in the absence of a demand for fueling fast reactors, while simultaneously maintaining an industrial ability to improve the know-how of reprocessing and recycling technology. The detailed information available on the nuclear program in France provides a benchmark that indicates the level of commitment needed to develop monorecycling. Notably, even with substantial investment, France is still decades away from achieving more advanced fuel cycles, putting into perspective the claims of advanced reactor developers that certain progress could take place on a shorter time frame. In the United States, industrial experience with reprocessing was short-lived and rather unsuccessful, as measured by the experience at the West Valley plant (shut down in 1972) and the Morris facility (determined to be inoperable in 1972). The amount of fuel reprocessed at West Valley over a period of 6 years was less than 1 percent of the amount of spent fuel that had been generated at that time by the U.S. fleet of LWRs. Therefore, the issue of plutonium storage or reuse is not a current issue for the U.S. commercial sector.

The default option for the existing U.S. spent fuel inventory remains the once-through nuclear fuel cycle, which is still not being fully implemented because of the political impasse over the Yucca Mountain geologic repository site in Nevada. No incentives presently exist for undertaking monorecycling in the United States, largely because of the high costs involved and the decreasing contribution by LWRs to the generation of electricity due to plant shutdowns; substantial challenges, based on past experience, for successful licensing and construction of spent fuel reprocessing and mixed oxide fuel fabrication installations; security and environmental concerns; and the abundance of natural uranium and uranium enrichment at relatively low costs for the foreseeable future.

Finding 1: Substantial, sustained investments to 2050 and beyond are required to develop technically complex advanced nuclear technologies and fuel cycle facilities and to enable potential commercial success. Notably, France has had a consistent vision on nuclear energy's role in its energy security for more than five decades, and as a result, the majority of France's electricity comes from nuclear power; however, after delaying development of fast reactors, it has yet to close the fuel cycle and is still decades from doing so. No matter what fuel cycle option the United States chooses—whether direct geologic disposal or a closed fuel cycle using advanced technologies—long-term vision and significant and sustained financial commitment will be required to execute it.

Finding 2: Continued use of the once-through fuel cycle for the existing U.S. light water reactor (LWR) fleet has several merits: (1) lower cost compared with any fuel cycle that involves reprocessing and recycling; (2) a reliable international market for nuclear fuel services from multiple suppliers (although that could be disrupted by international crises, such as war)[1]; (3) compatibility with the projected available uranium resources; (4) well-understood proliferation resistance of the entire fuel cycle; and (5) theft resistance of spent nuclear fuel. However, the once-through cycle remains incomplete in the United States because there is still no progress toward establishing an operating geologic repository for the spent fuel from nuclear power plants. Pursuing the monorecycling fuel cycle with existing LWRs in the United States would add cost to nuclear power generation but produce no significant benefits, given the projected abundant supply of natural uranium and uranium enrichment at relatively low cost for the foreseeable future.

2.2 DEVELOPMENT OF THE CURRENT GENERATION OF NUCLEAR POWER PLANTS AND SUPPORTING FUEL CYCLES

In the United States, starting in the late 1940s and continuing through the 1950s, the U.S. Atomic Energy Commission (AEC), along with utility and industry partners, spearheaded development and construction of the first U.S. nuclear power reactors. In particular, the world's first full-scale nuclear power plant devoted exclusively to

[1] The October 2020 agreement between the U.S. Department of Commerce and Rosatom (the Russian state nuclear energy corporation) allows Russia to continue to export enriched uranium to the United States, but it reduces the proportions from approximately 20 percent of U.S. demand to no higher than 15 percent from 2028 to 2040. The amendment also limits the natural uranium and uranium conversion services from Russia to an amount equivalent to no more than 5 percent of U.S. enrichment demand from 2026 to 2040.

peacetime uses for generating electricity was a demonstration pressurized water reactor,[2] a type of LWR, located at Shippingport, Pennsylvania, that reached criticality in December 1957 (DOE-NE, 2002). The 1960s and 1970s brought further development and construction of commercial LWRs for electricity generation, taking advantage of the technology and facilities developed for defense purposes, especially the capability, developed during the Manhattan Project, to enrich uranium. Additionally, the U.S. Navy's Naval Nuclear Power Program led developments in reactor technologies, including the pressurized water reactor that was a model for Shippingport. These nuclear power plants were promoted as relatively inexpensive and emission-free sources of electricity. As a result, nuclear power was seen at the time as becoming the major power source for electricity generation of the future. However, because the natural abundance of uranium ore was perceived as being potentially scarce, the AEC was concerned about the availability of sufficient material for both nuclear weapons and nuclear power programs, and sought an alternative to long-term sole reliance on uranium-fuel–based LWR technology.

As a result of this concern, at the outset of commercial nuclear power development, the AEC sought to develop a type of reactor that would have the potential of generating more fissile material than it consumed, hence the term *breeder reactor*. A plutonium breeder reactor requires a fast neutron spectrum, so thermal-spectrum LWRs cannot be used.[3] The main plutonium isotope, plutonium-239, is created by neutron capture in uranium-238, which is 138 times more abundant than uranium-235. To accumulate the initial plutonium for the first core of a breeder reactor, it is necessary to first recover the plutonium present at a low concentration (less than ~1 weight-percent) in the spent fuel discharged from LWRs. Therefore, from the beginning, a significant amount of R&D and demonstration worldwide has been focused on developing and maturing the technologies associated with (1) breeder reactor design, fueling, and operation, and (2) chemical recovery of the plutonium from irradiated LWR fuel. Both technologies—breeder reactors and reprocessing—turned out to be technically complex and economically challenging. After several decades of development, no country has yet operated an economic and reliable commercial breeder reactor.

In the late 1940s, the AEC authorized construction of the Experimental Breeder Reactor-I (EBR-I) in Idaho to investigate a technology that could breed fissile material (plutonium[4]) from fertile uranium-238. On December 20, 1951, EBR-I generated the first electricity from nuclear energy, although it was intended to be used for experiments and not as an electricity-generating reactor. On November 29, 1955, this reactor experienced a partial fuel meltdown during a coolant flow test. Despite this accident, EBR-I proved the principle of plutonium breeding in a fast reactor. The reactor was repaired and continued to be used for experimental purposes. It was deactivated in 1964.

Next, the Experimental Breeder Reactor-II (EBR-II), a 20-MWe (megawatts electric) unit, supplied electricity intermittently from 1964 to its closing in 1994. "The original emphasis in the design and operation of EBR-II was to demonstrate a complete breeder-reactor power plant with on-site reprocessing of solid metallic fuel" (Westfall, 2004). The original breeder cycle testing was conducted until 1969; subsequently, EBR-II was used for testing concepts for the proposed integral fast reactor.[5]

In 1970, the Clinch River Breeder Reactor project was authorized. The project's conception was to lead to development of liquid-metal breeder reactor technology for commercial electric-power generation in the United

[2] In addition to electricity generation, the Shippingport reactor was used to demonstrate "seed-and-blanket" technologies, in which neutrons from the reactor core "seed" were used to breed fissile material in a "blanket" of fertile material such as uranium-238 and thorium surrounding the core. One demonstrated configuration tested for using the reactor as a thermal breeder, but the reactor did not actually breed. As explained in the main text and next footnote, breeder reactors have almost always used fast neutrons. Breeding is theoretically possible in thermal molten salt reactors operating on the thorium/uranium-233 cycle if protactinium-233 is extracted (Nagy et al., 2008).

[3] Fast, or highly energetic, neutrons result, on average, in more neutrons produced per fission than slow, or thermal energy, neutrons. Having excess neutrons provides more neutrons for breeding fissile material from fertile material in breeder reactors. To make sure that neutrons are not slowed down, breeder reactors use metallic coolants, because metals such as lead and lead-bismuth eutectic do not slow down colliding neutrons appreciably, even after a neutron experiences several collisions. In comparison, water used as coolant in LWRs has a significant neutron slowing down, or moderating, effect even after just a few collisions between neutrons and hydrogen atoms in water.

[4] Plutonium consists of several isotopes, the main ones being Pu-238, Pu-239, Pu-240, Pu-241, Pu-242, and Pu-244. Both Pu-239 and Pu-241 have favorable characteristics for fuel breeding technology when they interact (i.e., fission) with high-energy neutrons. The isotopic makeup of plutonium is a function of the composition of the starting material makeup and irradiation conditions.

[5] The integral fast reactor would breed more fuel than EBR-II and would be distinguished by a nuclear fuel cycle that performs reprocessing via electrorefining (see Chapter 4) at the reactor site.

States and was intended to be a 350-MWe prototype and demonstration for this type of reactors. Continued escalation in the cost of the project (from an initial cost estimate of ~$400 million to more than $3 billion) and concerns about nuclear weapons proliferation eventually resulted in its termination in 1983 (Breeder Reactor Corporation, 1985).

In addition to the programs in the United States, several test, experimental, or prototype breeder reactors were built in other countries (MIT, 2018), including a commercial-scale reactor, France's Superphénix, which was connected to the grid in 1986 and permanently shut down in 1998.[6] Some of these reactors experienced operating difficulties, including some accidents that resulted in shutdowns (IAEA, 2004b). Today, only Russia operates two industrial-size fast reactors: the BN-600 and BN-800. Russia's experience with operating commercial-size fast reactors is described in Appendix H.

As noted above, recovery of plutonium from reprocessing spent LWR uranium oxide fuel is required to support the development and fueling of breeder reactors. To that end, several fuel reprocessing facilities have been built and operated, most notably in France, Russia, the United Kingdom, and the United States. The West Valley Demonstration Project (New York) was the first, and to date the only, commercial LWR fuel reprocessing plant in the United States.[7] Starting in 1966 with a potential capacity to reprocess 300 MT (metric tons) of spent LWR fuel annually, the facility eventually reprocessed about 640 MT in 6 years before shutting down in 1972 in response to new regulatory and environmental requirements. Required plant modifications were predicted to increase the reprocessing costs by an order of magnitude, and U.S. utilities chose not to enter into additional contracts at the increased price, given better prospects for natural uranium resources and delays in the development of fast reactor technology. As a result, the owner abandoned the plant, and West Valley became a nuclear waste cleanup site with an estimated remediation cost of $4.5 billion (GAO, 1977; von Hippel, 2007).

In 1974, India tested a nuclear explosive device using plutonium produced with technology obtained from Western supplies; this triggered a major shift away from reprocessing in U.S. nonproliferation policy in 1976.[8] This new policy was intended to discourage further adoption of technologies applicable for potential weapons production. It also effectively emphasized and encouraged global implementation of the once-through fuel cycle concept. The simplicity, favorable economics, and nonproliferation characteristics of the once-through fuel cycle facilitated its acceptance with the nuclear power industry in the United States and in many other countries.

As the commercial nuclear industry was developing in the United States, the Navy's Naval Nuclear Power Program selected thermal LWR technology to power its nuclear fleet. This selection and the program that developed were accompanied by a corresponding buildup of a supporting industrial base and associated expertise among the industry and regulators. The commercial nuclear industry, leveraging these economies of scale, also preferentially deployed LWRs in its power plants. The adoption of the once-through fuel cycle and LWR technology by the U.S. nuclear industry has resulted in low-enriched uranium oxide fuels becoming the single de facto standard nuclear fuel, thereby simplifying the fuel cycle. An efficient infrastructure emerged to support the use of a single fuel type, and considerable research has been performed to further improve the economics of the uranium oxide once-through fuel cycle. On the back end, understanding of spent fuel performance in storage and disposal has focused on uranium oxide fuels. The industry has taken advantage of the learning curve with uranium oxide fuels to establish a mature supply chain and well-understood waste behavior to reduce costs and improve performance.

From the late 1970s to today, the relatively low cost of uranium, the abundance of natural uranium, and the deployment of more economical methods for enriching uranium have all contributed to the viability of the once-through fuel cycle for LWRs, not only in the United States but also in most nuclear power–producing countries. Reprocessing of LWR fuel has continued in France and Russia with various degrees of commercial success, as illustrated by the ebb and flow of contractual agreements between reprocessing plant owners and utilities. A sum-

[6] The Superphénix plant started its operation 4 months before the Chernobyl accident, and the project was eventually terminated by governmental decision in 1997. Despite its complicated "political life," Superphénix provided a wealth of experience on construction and operation of an industrial-size prototype.

[7] Two additional commercial reprocessing facilities were constructed but never operated: General Electric's Midwest Fuel Recovery Plant at Morris, Illinois, which was completed but declared inoperable in 1970, and Allied General Nuclear Services, which began construction of a 1,500-tons/year facility in Barnwell, South Carolina, but was canceled in 1977.

[8] President G. Ford in 1976 first began the policy of "indefinite deferral of reprocessing."

mary of reprocessing activities in relevant countries is provided in Appendix H. For more information about the history of U.S. nuclear reactor and fuel cycle development, see Vine (2011).

2.3 COMMERCIAL NUCLEAR FUEL CYCLE OPERATIONS SUPPORTING LIGHT WATER REACTORS

The nuclear fuel cycle consists of the front end, which involves the preparation of the fuel; the service period, in which the fuel is used during reactor operations; and the back end, during which the spent fuel and nuclear waste is managed, stored, and disposed.

Figure 2.1 shows the once-through fuel cycle operations that are currently supporting the operation of most LWRs worldwide, as well as the steps for the monorecycle option being implemented in a few countries.[9] Transport of fuel or materials created by nuclear fission is shown by arrows between different activities or facilities. Transportation, an important element of any fuel cycle activity, is covered in greater detail in Chapter 5. Not shown in Figure 2.1 is the anticipated final disposal of spent fuel or high-level wastes in a geologic repository, because

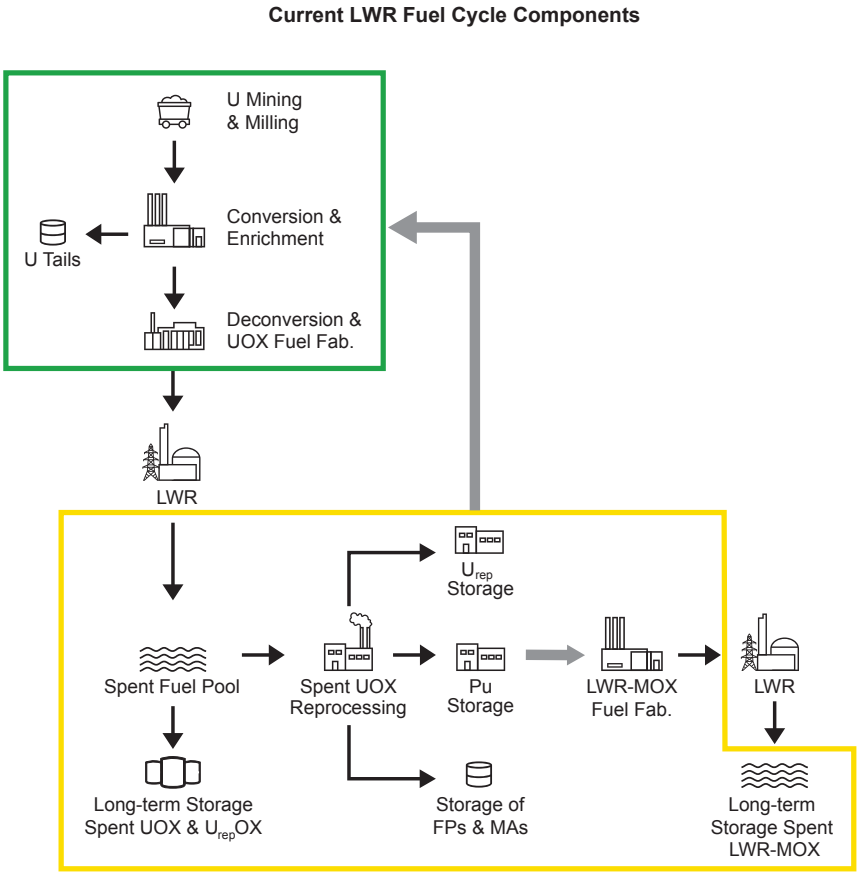


FIGURE 2.1 Today's LWR fuel cycle components—established processes and facilities in commercial operation.
NOTES: Front- and back-end facilities or operations are highlighted in green and yellow, respectively. FP = fission product; LWR = light water reactor; MA = minor actinides; MOX = mixed oxide; UOX = uranium oxide; Urep = reprocessed uranium.
SOURCE: Icons adapted from MIT (2011).

[9] Figure 2.1 is an oversimplification of the once-through fuel cycles that have been implemented in national programs. See Krahn et al. (2014) for a more recent assessment of the contemporary nuclear fuel cycle in the United States.

there are presently no such operating facilities for commercial spent nuclear fuel anywhere in the world, although repositories in Sweden and Finland may begin operations later this decade.

2.3.1 Front-End Operations: A Global Enterprise

The front-end technologies and facilities required to sustain LWRs include uranium mining and milling, conversion and enrichment, deconversion, and fuel fabrication. These technologies are well established and comprise a global network of nuclear fuel services. However, this global nature of the nuclear fuel supply chain can make the United States vulnerable to foreign disruptions, including war, as evidenced by the 2022 Russian invasion of Ukraine.

Uranium mining and milling: Since the 1970s, prospecting has discovered that uranium is significantly more abundant than previously assessed (NEA-OECD, 2006a). While 20 countries have uranium mines, in recent years, more than 50 percent of production of uranium comes from nine mines in four countries: Kazakhstan, Canada, Namibia, and Australia (WNA, 2021a). Using only the open, once-through fuel cycle, the world has sufficient uranium resources to meet the demands of nuclear power for the foreseeable future. In particular, according to the Nuclear Energy Agency's assessment, as of the end of 2019, the global recoverable uranium resources, based on a price of $130 per kilogram uranium, are sufficient for over 135 years, considering global requirements (NEA and IAEA, 2020).

The United States, once having a substantial domestic uranium mining industry, currently contributes only about 1 percent of the global uranium market because of the relatively low grade of uranium ores in most U.S. mines. Increasingly, since the late 1980s, U.S. commercial nuclear power production has relied on foreign-origin uranium for reactor fuel. In 2020, for example, the U.S. nuclear power sector purchased about 70 percent of its uranium from just four countries: Canada (22 percent), Kazakhstan (22 percent), Russia (16 percent), and Australia (11 percent) (EIA, 2021a).

Uranium conversion and enrichment: Chapter 1 introduced the fundamentals of conversion and enrichment. Here, the focus is on the status of global capacities for conversion and enrichment.

The four primary uranium conversion companies are Russia's Rosatom at about 38 percent, China's National Nuclear Corporation and Canada's Cameco at about 25 percent each, and France's Orano at about 8 percent (WNA, 2022b). As mentioned in Chapter 1, there is currently no operating conversion facility in the United States, although the Honeywell Metropolis Works facility plans to restart operations in 2023 (WNN, 2021d).

While 13 countries have some enrichment capabilities, approximately 90 percent of the enriched uranium is produced in seven countries by four companies: Tenex (Russia), Urenco (Germany, the Netherlands, the United Kingdom, and the United States), Orano (France), and China National Nuclear Corporation (WNA, 2020a). The only operating enrichment plant in the United States is owned by Urenco. Located in Eunice, New Mexico, this plant supplies almost one-third of U.S. commercial nuclear power industry requirements for uranium-235 enriched to less than 5 percent (WNA, 2021b).

Deconversion and fuel fabrication: "The enriched uranium is transported to a fuel fabrication plant where it is converted to uranium dioxide powder" (WNA, 2021c). This powder is then pressed and sintered into fuel pellets, which "are subsequently inserted into thin tubes known as fuel rods, which are then grouped together to form fuel assemblies" (WNA, 2021c). The number of fuel rods and exact dimensions of a fuel assembly vary and depend on the specific design of the reactor lattice (WNA, 2021c).

Domestic fuel fabrication is the only step of the fuel supply chain that is currently sufficient to meet the needs of the U.S. commercial power industry for uranium oxide fuel with less than 5 percent enrichment. Currently, three fuel fabrication plants are licensed by the U.S. Nuclear Regulatory Commission to produce low-enriched uranium fuel that is sold worldwide: Global Nuclear Fuel-Americas in Wilmington, North Carolina; Westinghouse Columbia Fuel Fabrication Facility in Columbia, South Carolina; and Framatome, Inc., in Richland, Washington.

2.3.2 Back-End Operations: Managed Storage

Assemblies loaded in an LWR typically generate energy for 4–6 years before the fuel is no longer suitable for power production. The assemblies are then discharged from the reactor and placed in an underwater storage and cooling facility, commonly referred to as a "spent fuel pool," located at the reactor site.

During the 1960s and 1970s, when the nuclear power plants were first deployed at large commercial scale, it was anticipated that spent fuel would be removed from the spent fuel pool after a few years of cooling and shipped to a reprocessing plant in order to extract the plutonium to fuel an expanding fleet of fast reactors. While the prospects for fast reactors fueled with plutonium dimmed progressively in the 1970s and 1980s, countries that developed reprocessing programs (with government support) with the intent of separating plutonium for fast reactors began to market the technology for management of spent fuel and monorecycling in LWRs instead. Commercial reprocessing services for LWR fuel became available in the United States (West Valley) in 1966 and in France (La Hague) in 1976, although as discussed above, the West Valley plant was shut down in the 1970s. In 1979, the Thermal Oxide Reprocessing Plant (THORP) for uranium oxide fuel began construction at Sellafield in the United Kingdom and started operations in 1994.[10]

Several Western European (France, Belgium, Germany, Italy, the Netherlands, Spain, and Switzerland) and Japanese utilities took advantage of the commercial reprocessing services available in France and the United Kingdom for an extended period of time. However, many of these utilities no longer use commercial reprocessing services, for a variety of reasons, including nuclear power phase-out policies and economic penalties deriving from the higher expense of reprocessing relative to spent fuel storage, the higher fabrication cost of plutonium-based mixed oxide fuels compared with uranium fuels, and the lack of prospects for plutonium reuse in fast reactors. Presently, France and Russia continue to reprocess, and Japan intends to do so in a facility located at Rokkasho. Between La Hague (France)[11] and Sellafield (United Kingdom),[12] ~50,000 MT of spent LWR fuel have been reprocessed.

Today, most of the approximately 30 countries with nuclear power programs neither reprocess spent nuclear fuel nor use mixed oxide fuels. Only France and Russia currently operate commercial-scale reprocessing facilities. China has one operating small-scale facility for reprocessing civilian nuclear fuel and one under construction. The countries that currently reprocess or have explored this technology in the past represent a large fraction of global nuclear energy generation. About 10 percent of the world's reactors are licensed to use mixed oxide fuel, but mixed oxide makes up only about 5 percent of the world's new nuclear fuel (WNA, 2017a). See Appendix H for a summary of reprocessing and recycling programs in China, India, Japan, Russia, and the United Kingdom, and Section 2.5 for a comparison of reprocessing programs and policies in France and the United States.

2.3.2.1 Management of Spent Fuel in the Once-Through Cycle

Spent fuel generated by a once-through fuel cycle is managed in interim storage until a final disposal site becomes available. In the United States, continued delays have required utilities to assume the responsibility for storing spent fuel for much longer than anticipated, including beyond the lifetime of the plants. Initially, utilities typically opted for reracking the spent fuel pools to accommodate larger numbers of assemblies. By the mid-1980s, it became apparent that space limitations would eventually require a significant fraction of the spent fuel inventory to be transferred out of spent fuel pools into dedicated interim storage systems.

Some countries built at-reactor systems for dry storage under inert conditions, while others built consolidated interim storage facilities, both wet (Sweden) and dry (Germany).[13] A variety of both cask- and canister-based dry technologies have been employed.

[10] The United Kingdom's earlier reprocessing facility was for Magnox fuels, which is not relevant to this section.

[11] Orano La Hague reprocesses ~1,100 tons of spent fuel per year from EDF, the world's leading nuclear operator, totaling ~40,000 tons of spent fuel reprocessed to date (Orano, n.d.).

[12] Reprocessing operations ended in November 2018 at THORP after 24 years of operation because of "a significant downturn in demand. The plant reprocessed 9,331 tons of spent fuel" (WNN, 2018).

[13] In the United States, two consolidated interim storage facilities located in Texas (Interim Storage Partners) and New Mexico (Holtec) are under active development; the Interim Storage Partners' facility received a 40-year license in September 2021, while the Holtec license application is presently under review.

See Chapter 5 for a discussion about the laws and regulations on spent fuel storage and the impact on disposal in a permanent repository.

2.3.2.2 Management of Fissile Products and Waste Streams Resulting from Spent LWR Fuel Reprocessing

When reprocessing is implemented, spent LWR fuel is first transported from the spent fuel pools located at the reactor sites to the pool(s) located at the reprocessing facility. In France, for example, spent uranium oxide fuel is shipped to the La Hague facility within 2 years following discharge from the reactor and is stored in the spent fuel pools located at La Hague for a minimum of about 5 years prior to reprocessing.

The main product streams from reprocessing operations typically consist of reprocessed uranium, reactor-grade plutonium, vitrified high-level waste[14] (primarily fission products, any residual plutonium, and minor actinides); fuel assembly hardware waste (cladding, grids, thimble tubes, nozzles, etc.); and low-level waste, including effluents. A main drawback of this approach is an accumulation of reactor-grade plutonium stockpiles (see Table 2.1). Storage of plutonium is expensive because of physical protection requirements. In addition, reuse of the plutonium after extended storage would likely require chemical separation of the americium-241 (half-life of 432 years) that builds up in the stored plutonium as the result of the decay of plutonium-241 (half-life of 14 years). Loss of fissile plutonium-241 and buildup of americium-241 degrade the reactivity of the fuel in thermal and fast reactors, and increase worker dose during mixed oxide fuel fabrication.[15]

To mitigate the buildup of the plutonium inventory, reactor-grade plutonium can be substituted for enriched uranium, and fuel rods containing a mixture of depleted or natural uranium oxide and plutonium oxide (i.e., mixed oxide fuel) can be fabricated and used in LWRs. Reprocessed uranium and separated plutonium have been recycled in several European and Japanese LWRs. Recycling of both plutonium and reprocessed uranium can result in natural uranium savings of up to ~20 percent. Recycled plutonium alone helps reduce up to 12 percent of the amount of natural uranium and enrichment work required for producing new fuel (NEA-OECD, 2021). This approach, which reduces the amount of natural uranium and enrichment required for producing new fuel, decreases the economic penalty associated with reprocessing. However, even if the cost of reprocessing is not included, "MOX [mixed oxide] fuel is more expensive than uranium fuel" (Bunn, 2021).

Plutonium inventories, as shown in Table 2.1, may result from technical and institutional factors that prevent keeping plutonium separation and reuse in LWRs in balance, or more generally from the fact that the market for plutonium for fueling fast reactors has not materialized.

Reprocessing: Commercial reprocessing of spent LWR fuel relies on the plutonium and uranium extraction (PUREX) process, an aqueous process for treating spent LWR fuel that was developed at Oak Ridge National Laboratory (Long, 1967). PUREX has more than 50 years of operational experience at commercial scale, during which time significant improvements in plant operations and safety have been incorporated (Poinssot, 2021). Presently, La Hague, where about three-quarters of all LWR fuel reprocessed worldwide has been treated to date, is the largest PUREX-based reprocessing facility for LWR fuel. For more information about the PUREX process and the French reprocessing program, see Chapter 4.

Reprocessed uranium recycle and storage: Recycling of reprocessed uranium prior to fuel fabrication has previously been implemented in several countries, notably in Belgium, France, Germany, Japan, the Netherlands, Sweden, and Switzerland (IAEA, 2007a, 2009). Presently, reprocessed uranium is not intended to be recycled, except in France, primarily for economic reasons, but that is subject to change if natural uranium costs increase

[14] According to a U.S. Nuclear Waste Technical Review Board fact sheet, "HLW [high-level (radioactive) waste] is vitrified by mixing it with a combination of silica sand and other glass-forming chemicals, heating the mixture to very high temperatures [approximately 1,150°C (2,100°F)] until it melts, and pouring the molten material into stainless steel canisters where it cools to form a glass" (U.S. Nuclear Waste Technical Review Board, 2017).

[15] Safety issues can also arise from storage of plutonium over extended periods; as an example, helium accumulation in the United Kingdom's packages of plutonium dioxide produced at THORP could lead to higher internal pressures and embrittlement of plutonium dioxide ceramics, such as mixed oxide fuel (Hyatt, 2020).

TABLE 2.1 National Holdings of Civil Separated Plutonium as of 2021

	Unirradiated Separated Plutonium (MT of Pu)	
Country	Held Domestically	Held Abroad
United States	49.3	0.0
France	79.4	<0.05
United Kingdom	116.1	0.0
Japan	8.9	37.2
Russia	63.3	0.0
China	0.04[a]	0.0[a]
India	0.4	0.0
Germany	0.0	0.0
Belgium	<0.05	0.0
Switzerland	<0.002	0.0
Sweden	0.0	0.834[b]

[a] Values as of December 31, 2016 (from China's most recent information circular as of March 2022).
[b] Value as of March 2014 (IPFM, 2014). Swedish separated Pu is held in the United Kingdom.
SOURCES: IAEA (2021c); IPFM (2021)

(EPRI, 2010a). Reprocessed uranium contains two additional uranium isotopes (uranium-232 and -236) that complicate reuse in enrichment and reactors.[16] Reprocessed uranium (U_{rep}) in LWRs can be recycled "via direct enrichment using centrifuge technology. In this case, U_{rep} experiences essentially the same process as natural uranium feedstock, albeit with some purification as well. U_{rep} is converted into UF_6 and then sent as soon as possible to the enrichment plant to avoid the buildup of ^{232}U daughters" (EPRI, 2010a). Reenrichment of reprocessed uranium must be performed in a dedicated cascade to avoid cross contamination by uranium-232.[17] In the absence of a dedicated cascade, reenrichment of reprocessed uranium can be obtained by downblending, using some of the existing stockpiles of excess defense-related enriched uranium; this approach avoids the buildup of uranium-236 and its associated neutronic penalty (EPRI, 2010a).

Because reprocessed uranium may require overenrichment in uranium-235 to compensate for the neutron-absorbing uranium-236, and because natural uranium resources are available at reasonably low cost, fuel economics favor the use of fresh rather than recycled reprocessed uranium in existing LWRs. As a result, some countries have chosen not to recycle reprocessed uranium and are storing it until its use becomes economically competitive.[18] Countries that have exercised the fuel cycle option to reenrich and recycle reprocessed uranium in an LWR have successfully managed the radiological aspects of uranium-232 associated with handling reprocessed uranium during conversion, reenrichment, fuel fabrication for recycle, transportation, and fuel loading during reactor operations. Spent reprocessed uranium oxide is not presently scheduled for further reprocessing[19] and will require storage until a decision is made to either reprocess or dispose of it in a geologic repository, when one becomes available. If a decision is made to not recycle reprocessed uranium and its use is no longer considered economically advantageous, the stored inventory of reprocessed uranium could be disposed of as low-level or Greater-than-Class-C (GTCC) waste (IAEA, 2007a). (See Appendix D for waste classifications.)

[16] Buildup of uranium-232 decay products over time causes both contamination and radiological hazards, resulting from the intense beta and gamma radiation of its short-lived daughter products (lead-212, bismuth-212, and especially thallium-208). Uranium-236 is a parasitic neutron absorber that requires an increase in enrichment of uranium-235 of ~0.5 percent.

[17] Such a facility is available in Russia and planned in France.

[18] Most of the reprocessed uranium remains in storage, though its conversion and reenrichment (in the United Kingdom, Russia, and the Netherlands) has been demonstrated, along with its reuse in fresh fuel. In Belgium, France, Germany, and Switzerland, over 8,000 MT of reprocessed uranium have been recycled into nuclear power plants (WNA, 2020b).

[19] Compared with spent uranium oxide, spent reprocessed uranium oxide contains ~4 times more uranium-232 and ~3 times more uranium-236, and accordingly presents a greater radiological hazard and has a more consequential neutron absorber-to-fissile content ratio.

Plutonium recycle and storage: Plutonium can be recycled in an LWR by blending it with natural or depleted uranium to form a mixed uranium-plutonium oxide fuel, or LWR-MOX. Fabrication of MOX fuel was pioneered for breeder reactor fuel and adapted for LWR fuel as an alternative to low-enriched uranium oxide. To use MOX, existing LWRs need to make some modifications, such as adding more control rods and amending their safety analyses and licenses (EPRI, 2009a). The core loading of LWR-MOX (with typically 8–10 percent plutonium) is limited to less than 50 percent, though most LWRs operate with a core loading of around one-third.[20] Notably, utilities in the United States are not using MOX fuel, and during the recent development of a potential MOX facility at the Savannah River Site, use of this fuel by utilities was conditional on the U.S. government providing financial support.

Irradiation of MOX fuel to the burnup targets of interest in LWRs typically limits the fuel to monorecycling because LWR-MOX fuel is significantly different from conventional uranium oxide fuel from a neutronic point of view. The fissile quality of the plutonium (i.e., the relative fraction of fissile plutonium [plutonium-239 and -241] to total plutonium) is lower in the plutonium recovered from spent LWR-MOX. Box 2.1 illustrates the impact of monorecycling on the buildup of plutonium, americium, and curium isotopes in the spent fuel. It would then be necessary to increase the plutonium content for a second recycling in order to compensate for the decrease in fissile quality. Deterioration of the safety parameters[21] beyond plutonium concentrations of ~12 percent becomes a safety barrier to multirecycling of plutonium in present-day LWRs.[22] As a result, enriched uranium has to be added to the fuel for such a potential option.

From the perspective of spent fuel assembly volume, each spent MOX assembly is obtained by reprocessing seven spent uranium oxide assemblies, thereby significantly reducing the number of spent fuel assemblies to be stored. Therefore, it is also a more compact approach for interim storage of the Pu itself. However, a fair comparison of the impacts of waste storage requires accounting for all the waste streams generated, including high-level waste and reprocessed uranium. In addition, because of waste generated during reprocessing and MOX fuel fabrication, low-level waste volumes, including GTCC wastes (similar to intermediate-level waste), increase compared with uranium oxide storage.

Moreover, in addition to the handling, storage, and disposal technologies developed for spent uranium oxide (which are applicable to spent MOX), spent MOX fuel management must take into account decay heat, potential criticality safety, and radiation source terms. These additional considerations are required because the decay heat generation of spent LWR-MOX decreases more slowly than that of spent uranium oxide (see Figure 2.2). Therefore, wet storage of spent LWR-MOX assemblies is preferred over dry storage. Additionally, the fissile inventory and minor actinide content are significantly larger for spent LWR-MOX than for spent uranium oxide.

As discussed above, monorecycling of the plutonium recovered from spent LWR fuel by reprocessing has been implemented to avoid the economic penalties associated with storage of the extracted plutonium. Otherwise, there is little technical rationale for LWR spent fuel reprocessing and monorecycling, given the modest uranium resource savings and larger economic penalties compared with direct geologic disposal. However, some countries justify LWR spent fuel reprocessing as a means of separating plutonium for use in future fast breeder reactors, as well as establishing and maintaining the technological know-how for developing and operating reprocessing facilities. In any case, monorecycling would appear to be an interim strategy at best. Alternatively, countries that plan to pursue fast breeder reactors could defer LWR spent fuel reprocessing until there is a realistic prospect for use of the resulting separated plutonium to avoid the cost and security liabilities of long-term plutonium storage.

High- and intermediate-level waste storage and low-level waste disposal: Vitrified fission products/minor actinides (i.e., high-level waste [HLW]) and compacted fuel/process hardware (i.e., intermediate-level waste [ILW][23]) are stored in containers in a dry environment. While estimates vary, Bunn et al. (2003) cites British Nuclear Fuels Limited (BNFL)

[20] Some LWRs designed for 100 percent MOX are under construction (EPRs, Õma Nuclear Power Station in Japan), and others are capable of operating with 100 percent MOX (Palo Verde Station in the United States).

[21] These safety parameters are control rod reactivity worth, boron reactivity worth, moderator void, and temperature coefficients.

[22] *Present-day LWRs* refers to LWRs with conventional lattices. For example, a conventional 17×17 pressurized water reactor (PWR) lattice, such as used in many of today's PWRs, is characterized by a moderator (water)-to-fuel ratio of 2:1. Multirecycling of plutonium will typically require a higher moderator (water)-to-fuel ratio, and therefore will require different fuel assembly geometries and reactor internals.

[23] The United States does not have an ILW category, but it is similar to Greater than Class C waste. ILW is used here to provide an international context to the estimate of LLW generated by nuclear power plants.

BOX 2.1
Plutonium and Minor Actinide Contents of Spent MOX Versus Spent UOX Fuel

To illustrate the impact of monorecycling on the buildup of Pu isotopes and two important minor actinides, ^{241}Am and ^{244}Cm, assume that seven spent pressurized water reactor (PWR) uranium oxide (UOX) fuel assemblies are reprocessed to recover enough Pu to fabricate one new PWR-MOX (mixed oxide) assembly with an initial Pu content of ~8.65 percent. The table below provides a comparison of the contents in Pu isotopes and minor actinides in the seven reprocessed spent UOX assemblies and in the discharged spent PWR-MOX assembly. Fuel burnup (~50 GWd/MTU) and cooling time (~3 years) are similar for both types of spent fuel. For thermal reactors, such as LWRs, the even isotopes of Pu (-238, -240, and -242) do not contribute significantly to fissions; on the other hand, the fissile fraction of the Pu (i.e., the ratio (^{239}Pu+^{241}Pu) / (total plutonium)) is a measure of the fissile Pu quality for thermal reactors. As shown in the table below, total Pu content decreases by 27.6 percent and fissile Pu decreases by 41.4 percent. Increases in ^{241}Am (11 percent) and ^{244}Cm (55 percent) are also observed.

TABLE Mass Balances Normalized to One Discharged (Spent) MOX Assembly (Burnup = 50 GWd/MTU; Cooling: 3 years)

Nuclide(s)	Spent UOX, 7 assemblies [kg]	Spent MOX, 1 assembly [kg]	Column 2 – Column 3 [kg]	Difference [%]
^{238}Pu	1.15	1.10	–0.05	–4.3
^{239}Pu	20.80	11.00	–9.80	–47.0
^{240}Pu	9.28	8.70	–0.58	–6.2
^{241}Pu	5.12	4.20	–0.92	–18.0
^{242}Pu	2.88	3.40	+0.52	+18.0
Total Pu	39.23	28.40	–10.83	–27.6
^{239}Pu + ^{241}Pu	25.92	15.20	–10.72	–41.4
^{241}Am	0.99	1.10	+0.11	+11
^{244}Cm	0.029	0.045	+0.016	+55

NOTE: MOX = mixed oxide; UOX = uranium oxide.
SOURCE: Adapted from Figure 4-5 in EPRI (2010b).

and Compagnie générale des matières nucléaires (COGEMA) estimates, which indicate approximately 0.12 m^3/tHM (cubic meters per ton of heavy metal) of HLW. For ILW, BNFL's estimate is 0.8 m^3/tHM and COGEMA's is 0.35 m^3/tHM. Bunn et al. (2003) also state an estimate for low-level waste (LLW) of 2.8 m^3/tHM, which is substantially greater in volume than HLW and ILW combined. LLW can be disposed of in near-surface disposal sites, but doing so would require additional costs. As to the costs, when BNFL was reprocessing, it had permission from the UK government to negotiate substitution agreements with customers, in which a customer would agree to take back a larger amount of HLW in exchange for the reprocessing company handling disposal of the LLW. Shipping LLW back to the customer could be problematic given its large volumes. Bunn et al. (2003) estimate that LLW handling and disposal could add several tens of dollars per kilogram of spent fuel, which is on the order of 1 percent of the cost of reprocessing (Bunn, 2021).

Atmospheric and gaseous effluents: The release and dilution of some gaseous and liquid radionuclide (especially helium-3, krypton-85, and iodine-129) effluents have raised concerns about their impacts on human health and the environment. The French industry is working on tailored ceramics to confine specific radionuclides, such as iodine (CEA, 2009). According to a European Commission study in 2000, the radioactive discharges from La Hague and Sellafield were significantly greater than those from other nuclear power plants in Europe, but significantly smaller than those from the fertilizer (use of phosphogypsum[24] during the production of phosphoric acid) and petroleum[25] (pumping of oil and gas from the continental shelf in the North Sea) industries (Betti et al., 2004).

[24] Radionuclides contained in phosphogypsum are mainly uranium, thorium, and radon, and their decay products.
[25] Radionuclides contained in petroleum are radon and its decay products.

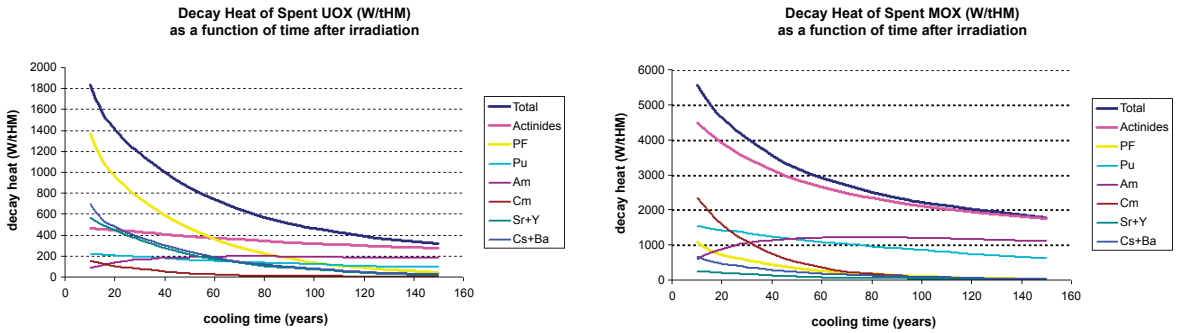


FIGURE 2.2 Decay heat of spent uranium oxide (UOX) (left) and spent mixed oxide (MOX) (right) irradiated to the same burnup (50 GWd/MTU).
NOTES: The two plots have different y-axis scales. The decay heat of the spent MOX is three times (after 10 years) to five times (after 150 years) higher than spent UOX.
SOURCE: EPRI (2010b). Used with permission from the Electric Power Research Institute, Inc.

2.3.3 Managed Storage

Independently of the option chosen for the LWR fuel cycle (once-through or monorecycling), the back-end endpoint of all commercial activities is, at the present time, *storage*. Spent fuel is stored as spent uranium oxide fuel, spent mixed oxide fuel, or spent reprocessed uranium oxide fuel, in either wet or dry environments, and either onsite with the reactor or at away-from-reactor sites. Interim storage of these spent fuel assemblies will be required until either a geologic repository is available for direct disposal or a decision is made to reprocess them. However, in the latter case, even if the separated fissile materials are used as feed material for a more advanced fuel cycle involving multirecycling, there will be spent fuel with low-residual-fissile content, which will not be worth reprocessing and will therefore require long-term management and disposal.

Product and waste streams from reprocessing, including plutonium, reprocessed uranium, vitrified high-level wastes, and low-level wastes (including GTCC waste), are in storage either at the reprocessing plant or in utilities' facilities. Product materials, which are recovered with the intention of recycling in LWRs or in more advanced fuel cycles, may also require long-term management if they cannot be readily utilized as fuel.

Given that presently neither geologic repositories for spent fuel nor commercial-scale advanced reactors are operating around the world,[26] back-end operations and implementation decisions have largely been driven by the need to ensure safe storage of the material containing fissile elements, minor actinides, and fission products. For this reason, back-end operations have been referred to as "managed storage."

2.4 COMPLETING THE LWR FUEL CYCLE

Interim storage, the final storage step in the management of spent fuel prior to final disposal, is presently the de facto endpoint for all nuclear power operations worldwide, but it does not presuppose or preclude any other endpoint for the spent fuel. Figure 2.3 depicts the back end of a fuel cycle, where it is assumed that interim storage is no longer the ultimate activity. Two additional options, referred to as "geologic repository" and "fast reactors" with recycling of plutonium and minor actinides, are shown in the figure.

2.4.1 Geologic Repository

As further discussed in Chapter 5, all fuel cycle options require a geologic repository. There is broad agreement in the scientific community that deep geologic disposal, if well-designed and executed, constitutes a safe option

[26] The two current exceptions are the Russian BN-600 and -800 reactors in Beloyarsk, and the Chinese HTR-PM reactor at Shidao Bay. Starting in 2020, the BN-800 reactor is being fueled with reprocessed uranium-plutonium fuel; the other two reactors (BN-600 and HTR-PM) use uranium-235; see Appendix H for more details.

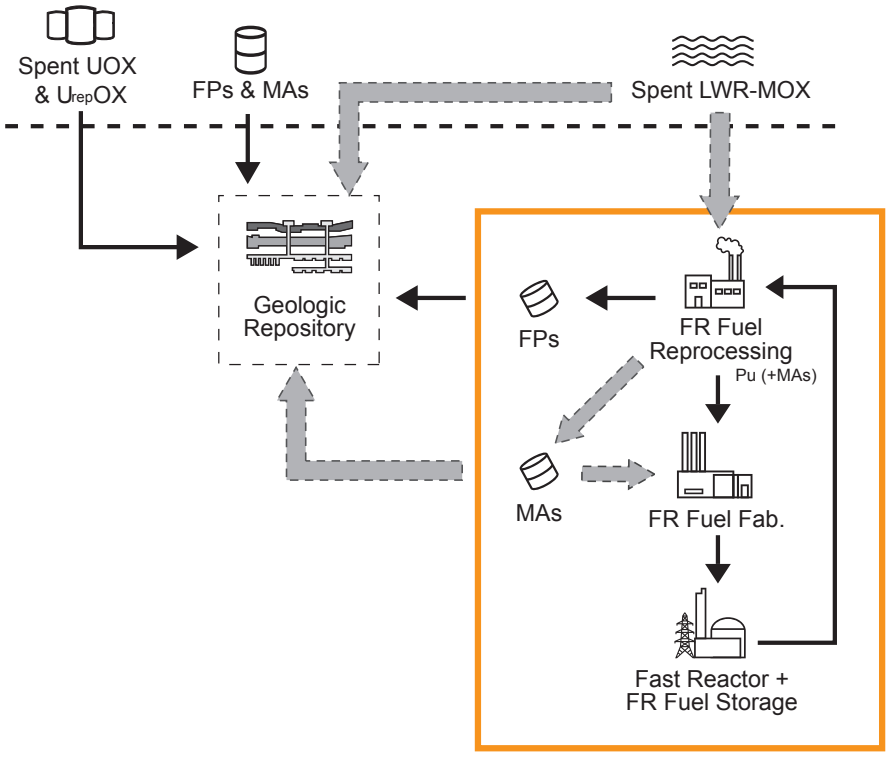


FIGURE 2.3 Back end of the fuel cycle, depicting the options of geologic repository, shown in gray, and of fast reactors recycling plutonium and minor actinides, shown in orange.
NOTES: The solid arrows represent essential steps, whereas the dotted arrows represent potential options. The geologic repository is shown in a dotted-line box to indicate that it is hypothetical, as no such facility currently exists. FP = fission product; FR = fast reactor; LWR = light water reactor; MA = minor actinides; MOX = mixed oxide; UOX = uranium oxide; $U_{rep}OX$ = reprocessed uranium oxide.
SOURCE: Icons adapted from MIT (2011).

for most forms of spent fuel and HLW generated by nuclear power plants and the nuclear fuel cycle (NEA-OECD, 2020). "The safety case for an HLW repository requires extensive R&D (regarding site suitability and waste packaging, for example) because the final selection of a site and disposal steps" (EPRI, 2010b) (emplacement, monitoring, retrievability, etc.) are expected to be challenged by both technical and societal issues. Although there is no geologic repository for commercial HLW in operation at this time,[27] Finland, Sweden, Canada, and France have made substantial progress toward licensing and could be operating repositories before the end of this decade. In Finland, operations at the Onkalo repository facility are expected to begin in 2024 or 2025, according to Posiva, the company behind Onkalo (*The Economist*, 2022). Sweden is just a few years behind, with its own repository at Forsmark. In France and Canada, repository facilities could begin operation in ~2035 (Bure, France)[28] and in 2040–2045 (Canada, site to be selected by 2024) (Dalton, 2022; WNN, 2022a).

2.4.2 Fast Reactor Technology

Fast reactors can potentially enable nearly full recovery of the energy contained in natural uranium resources by converting the fertile uranium-238 that makes up more than 99 percent of natural uranium into fissile plutonium-239. Plutonium recovered by reprocessing spent LWR uranium oxide fuel can be used to provide the initial

[27] In the United States, the Waste Isolation Pilot Plant is an operational geologic repository for defense-generated transuranic wastes.
[28] Estimate as of 2021.

plutonium inventory for fast reactor fueling, because, as discussed in Chapter 4, fissions induced in the fissile plutonium isotopes (-239 and -241) create neutrons in sufficiently large numbers to both maintain power production and produce more fissile material (plutonium-239) than is consumed. In an ideal breeder reactor system, no additional fissile material other than uranium-238 is needed from external sources. However, realizing such a system in practice will be challenging and will require multiple cycles of reprocessing of spent fast reactor fuel and fuel refabrication, as illustrated in Figure 2.3 (see Chapter 4 for additional details).

Historically, the most common design has been the sodium-cooled fast reactor. The largest operating fast reactor is presently the Russian BN-800 (2100 MWth [megawatts thermal]), which began operating in 2015 with a hybrid core consisting of both highly enriched uranium and mixed oxide fuel. See Chapters 3–5 for more details on sodium-cooled fast reactor technology.

Reprocessing of fast reactor mixed oxide fuel: Fast reactor spent fuels will have higher fissile concentrations than LWR fuels and potentially higher burnups. Technologies for reprocessing LWR spent fuels, such as PUREX, can generally be used for reprocessing fast reactor spent fuels, but some modifications are required to address criticality concerns and higher radioactivity levels.[29] In addition, pyrochemical methods are being developed in a few countries as integral parts of the refueling/waste management system for fast reactors, especially for nonoxide fuels. These methods would permit the treatment of different types of radioactive fuels with high plutonium content. In the longer term, fuel cycle applications specifically related to advanced reactor concepts may favor the use of pyrochemical processes (IAEA, 2008). These technologies are covered in Chapter 4.

Fast reactor fuel fabrication: Fuel fabrication will require more extensively shielded facilities compared with the glove box facilities in use for fabricating today's LWR and LWR-MOX fuels. Fuel fabrication is covered in greater detail in Chapter 4.

2.5 NUCLEAR FUEL CYCLE POLICIES: A COMPARISON BETWEEN THE UNITED STATES AND FRANCE

Given the history of the development and deployment of LWR technology and its fuel cycles, it is worth comparing how different energy markets and policy contexts result in different nuclear strategies. For this purpose, the United States and France are compared; while they have the two largest installed nuclear power capacities in the world, they have chosen different approaches to the nuclear fuel cycle.

2.5.1 The United States

The United States operates the largest nuclear fleet of power reactors in the world. At the end of 2021, 93 reactors—of a dozen different models of the pressurized and boiling water reactor types—were operating with a combined generation capacity of about 95,464 MWe (EIA, 2022). Reactor power uprates and high-capacity usage rates have contributed to nuclear power plants maintaining a consistent portion of about 20 percent of total annual U.S. electricity generation from 1990 through 2019.

[29] During reprocessing, the driving safety concerns are keeping the aqueous and organic solutions containing the radioactive, heat-generating nuclides at low-enough temperatures, and especially maintaining subcriticality of the aqueous and organic solutions containing fissile materials. Maintaining subcriticality involves some creativity with component geometry (e.g., annular "cylinders" instead of regular cylinders to maximize neutron leakage, and subsequent capture of the neutrons by neutron absorbers located on the inside and outside peripheries of the annular cylinder). A plant optimized to handle spent LWR fuel with a "low" fissile content could be reconfigured to reprocess other fuels. For example, La Hague has reprocessed spent LWR-MOX and fast reactor fuel, which contain high levels of fissile nuclides, using "dilution," in which a spent LWR-MOX assembly is reprocessed in a string of spent uranium oxide (UOX) assemblies with a UOX-to-MOX ratio on the order of 10 to allow the plant to stay within its safety boundaries. This approach relies on having an abundance of spent UOX fuel to reprocess with the other fuel. A facility dedicated to the reprocessing of spent MOX (LWR-MOX, enriched uranium MOX, fast reactor MOX) would require substantially greater redesign of equipment (pulse columns, centrifugal contactors, piping and container geometry, use of neutron absorbers, etc.) compared with a facility for reprocessing only spent UOX fuel.

U.S. policies on reprocessing and recycling technologies: As mentioned in Section 2.2, abandonment of the commercial reprocessing of spent nuclear fuel in the United States in the mid-1970s, due to proliferation and economic concerns, set the path of U.S. policy and its nuclear industry into a once-through fuel cycle. While the government ban on reprocessing was lifted in the early 1980s, freeing utilities to pursue reprocessing as long as they paid for it, U.S. commercial reprocessing was not economically competitive and never resumed. In 1993, to try to dissuade additional countries from separating plutonium from spent fuel, the Clinton administration reaffirmed the 1970s policy by stating that "the United States does not encourage the civil use of plutonium and does not itself engage in plutonium reprocessing" (White House, 1993) but committed to meeting its nuclear energy cooperative agreements with Japan and countries in Western Europe that reprocess or use reprocessed plutonium (Andrews, 2008).

In the early 2000s, however, the George W. Bush administration undertook another policy shift and invested in R&D for advanced reprocessing methods that would not completely separate plutonium from other elements, with the view that this might be more resistant to proliferation. The U.S. Department of Energy (DOE) provided research funding for a new program called the Advanced Fuel Cycle Initiative (AFCI) "to develop fuel cycle technologies for Generation IV reactors including reprocessing and using fast neutron reactors to transmute long-lived components of wastes" (WNA, 2021b). The Energy Policy Act of 2005 (Public Law 109-58 § 953) codified an objective of AFCI to "evaluate proliferation-resistant fuel recycling and transmutation technologies that minimize environmental and public health and safety impacts as an alternative to aqueous reprocessing technologies."

Moreover, in 2006, the George W. Bush administration launched the Global Nuclear Energy Partnership (GNEP), which proposed that the United States work with other developed nuclear power states that have existing fuel cycle capabilities to "develop proliferation-resistant recycling technologies and provide nuclear fuel to developing countries that promised not to engage in enrichment and reprocessing activities" (WNA, 2021b). When the Obama administration began in 2009, it canceled these aspects of GNEP "because it [was] no longer pursuing domestic commercial reprocessing, which was the primary focus of the prior Administration's domestic GNEP program" (DOE, 2009a).

In 2008, in response to DOE's GNEP program and industry interest in potential commercial-scale reprocessing and recycling, the U.S. Nuclear Regulatory Commission (U.S. NRC) started down the path of a possible rulemaking for updating the licensing basis for such facilities (U.S. NRC, 2011, 2013b). From 2013 to 2016, the U.S. NRC staff focused its analysis "on assessing the quantitative risk associated with reprocessing facility accidents" (U.S. NRC, 2021a). In 2016, the U.S. NRC staff observed that "industry interest in building and operating a commercial spent fuel reprocessing facility had declined" and consequently suspended its work on this activity (U.S. NRC, 2021a). In March 2020, the U.S. NRC staff sought public comment on whether it should resume the rulemaking or terminate it.

In May 2020, the American Nuclear Society (ANS) and the Nuclear Energy Institute responded by requesting that the U.S. NRC renew its rulemaking activity because of the renewed interest among several vendors in advanced reactor technologies that may benefit from reprocessing and recycling to achieve their full potential. Also, the ANS stated that "the lack of an efficient, technically robust, and technology inclusive regulatory foundation for reprocessing and recycling is a barrier to innovation in advanced reactor designs" (U.S. NRC, 2021b). In July 2021, the U.S. NRC filed notice in the *Federal Register* that it was formally suspending its efforts on potential rulemaking in this area. The U.S. NRC concluded that "in addition to using fresh fuel obtained from enrichment and fabrication, some advanced reactor designs have the capability to eventually source their fuel from the spent fuel of other reactors, but there was limited interest in pursuing reprocessing activities in the near future (within 10 to 20 years)," and that it was not worth the cost to the U.S. NRC to continue rulemaking in light of this limited interest (U.S. NRC, 2021b).

One outcome of the decisions since the 1970s by the U.S. government and utilities to not pursue commercial reprocessing is that, unlike France and several other countries, the United States did not generate a large stockpile of commercial plutonium, with its attendant security and safety risks. Given that the United States has experienced significant challenges over the past 30 years in seeking to dispose of its stockpile of roughly 50 MT of surplus military plutonium (see below), the addition of a comparably sized commercial stockpile would have greatly increased the disposal burden.

MOX fuel fabrication facility experience: The United States attempted to develop a capability for producing MOX fuel with separated military plutonium that the United States determined to be excess to defense needs. The MOX Fuel Fabrication Facility (MFFF), originally estimated in 2002 to cost $1 billion, was to be the central element of the National Nuclear Security Administration's (NNSA's) Plutonium Disposition Program and was initiated to "dispose" of at least 34 MT of excess weapons plutonium by using it in MOX fuel for commercial nuclear power reactors (Holt and Nikitin, 2017). The effort began in 2001 with Duke, COGEMA, Stone & Webster submitting a request to the U.S. NRC to construct the MFFF at the Savannah River site in South Carolina. The U.S. NRC granted a construction authorization in March 2005 (effective for 10 years) with ground breaking commencing approximately 2 years later. In 2014, the U.S. NRC was requested by the applicant (now CB&I Areva MOX Services, LLC) to extend the construction authorization. The request was granted, and the construction authorization was extended from the original March 2015 expiration date to March 2025 (U.S. NRC, 2020g). In October 2018, DOE issued a notice of termination of the construction contract for MFFF to CB&I Areva MOX Services, noting that the MFFF was "significantly behind schedule," and "would cost significantly more than previously projected" (Perry, 2019). NNSA is presently pursuing a dilute-and-dispose approach to dispositioning surplus weapons plutonium that would not use the plutonium as commercial reactor fuel (NASEM, 2020).

Economically competitive electricity markets and their effect on nuclear power in the United States: In the early 2000s, the Massachusetts Institute of Technology (MIT) published two major reports examining potential changes for the future of nuclear energy in the United States. The 2003 MIT report *The Future of Nuclear Power* observed that "if in the future carbon dioxide emissions carry a significant 'price,' nuclear energy could be an important—indeed vital—option for generating electricity," and advised that "the nuclear option should be retained, precisely because it is an important carbon free source of power that can potentially make a significant contribution to future electricity supply" (MIT, 2003). The update of this MIT study, published in May 2009, concluded:

> After five years, no new plants are under construction in the United States and insufficient progress has been made on waste management. The current assistance program put into place by the 2005 EPACT [Energy Policy Act of 2005] has not yet been effective and needs to be improved. *The sober warning is that if more is not done, nuclear power will diminish as a practical and timely option for deployment at a scale that would constitute a material contribution to climate change risk mitigation* (emphasis added). (MIT, 2009)

The past decade has provided confirmation of MIT's "sober warning." From 2013 through April 2021, 12 U.S. reactors were permanently shut down, and through the mid-2020s, seven more are slated for closure (Holt, 2021b). Increasing awareness of, and concerns about, climate change among the U.S. public and political leaders could have benefited social and political acceptance of nuclear power as a low-carbon alternative for baseload generation of electricity. However, the economic competitiveness of nuclear power in the form of LWRs has continued to degrade. Low-cost natural gas, production-credit incentives to renewables, and the lack of any carbon fee have limited the economic and political incentives for pursuing construction of new nuclear plants with large capital costs and potentially large financial risks. In addition, the combination of intermittent renewable and flexible electricity plants are progressively displacing future installed capacity and production of baseload generation technologies, which include LWRs. Moreover, about half of the U.S. nuclear power plants are in deregulated electricity markets and face daily economic competition from more cost-effective power sources. As a result, the U.S. Energy Information Administration projects that nuclear power's share of electricity generation will be reduced to about 11 percent by 2050 (EIA, 2021b). However, unanticipated market disruptions can quickly change the economic picture for nuclear energy, and the energy system more generally, as evidenced by the 2022 Russian invasion of Ukraine. These issues will be addressed in greater detail in the parallel National Academies study, Laying the Foundation for New and Advanced Nuclear Reactors, which is expected to be released in early 2023.

Past U.S. Department of Energy proposed justification for reprocessing and advanced fuel cycles: Around 1990, a new justification for reprocessing—the benefit of waste disposal—was promulgated for the U.S. nuclear

power development program. At the request of the U.S. Secretary of Energy, Admiral James Watkins, several organizations, including the Electric Power Research Institute (EPRI) and the National Academies addressed the question "Would the benefits to radioactive waste disposal justify processing of existing spent LWR fuel and deploying liquid metal fast reactors to consume the separated transuranic elements?"

Results from these studies tended to reach similar conclusions:

- According to EPRI, adoption of a process-before-disposal policy for current spent fuel would accrue minimal benefits. "The policy would likely incur a large cost penalty, encounter major institutional difficulties, multiply licensing difficulties, and amplify political and public opposition to the nuclear power program as a whole" (EPRI, 1991).
- According to the 1996 National Research Council report on separation and transmutation (S&T) (National Research Council, 1996):
  - A sustained and carefully focused program of R&D over the next decade "should focus on the factors that strongly influence fuel-cycle economics, especially the costs of reprocessing spent LWR fuel, minimization of long-lived radionuclides to secondary wastes in the reprocessing cycle, and on the need to reduce the possible increase in proliferation risks that could result from the commercial use of plutonium in recycled fuels."
  - "The current policy of using the once-through fuel cycle for commercial reactors, with disposal of the spent fuel as HLW, should be continued."
  - "None of the S&T system concepts reviewed eliminates the need for a geologic repository. DOE should continue efforts to develop a geologic repository for spent LWR fuel."

Based on congressional allocations and administration directions, DOE and its predecessor agencies have, for decades, conducted research on advanced reactor technologies, which differ significantly from existing commercial nuclear plants with respect to both the nature of the design and power capacity per unit. Since the mid-1980s, because of inconsistent or opposing perspectives on the role of nuclear energy for the United States, project outcomes have too often been limited to publication of comprehensive reports repetitively assessing these technologies, and rarely have proceeded to the construction of facilities where new concepts could be tested and evaluated at engineering scale.

DOE represents the U.S. government in the Generation-IV International Forum (GIF), which provides a framework for international cooperation in R&D for the next generation of nuclear energy systems, as further detailed in Chapter 3. This R&D collaboration has been ongoing for two decades and is expected to aid national and international progress toward the realization of such systems. Chapter 3 provides details on DOE's current advanced nuclear energy and associated fuel cycles programs.

Geologic repository: For the once-through fuel cycle, the major gap is the lack of a permanent geologic repository. As detailed in Chapter 5, the United States presently lacks a strategy for licensing and operating a geologic repository for commercial spent fuel.

Bottom line: U.S. energy policy has been in flux because of short-term oscillations in political support and leadership. Whatever choices are made about advanced nuclear fuel cycles, U.S. utilities will continue to operate LWRs as long as they remain cost-competitive over the next few decades. Seven more LWR units are expected to shut down by the end of 2025. Two new reactors, Vogtle 3 and 4, are scheduled to be in service in the first quarter of 2023 and by the fourth quarter of 2023, respectively, based on information from August 2022 (NEI, 2022).

In the *Annual Energy Outlook 2021*, the U.S. Energy Information Administration notes that "renewable energy incentives and falling technology costs support robust competition with natural gas as coal and nuclear power decrease in the electricity mix" (EIA, 2021b). Renewable electric generation is projected to meet an increasing share of additional demand. The report continues, "As the share of natural gas-fired generation remains relatively flat, and as the contribution from the coal and nuclear fleets drops by half, the renewables' share of the electricity generation mix more than doubles from 2020 to 2050" (EIA, 2021b).

2.5.2 France

France relies on nuclear generation for 70–75 percent of its electricity supply—the highest of any nation. This position results from implementing a deliberate national energy policy to prioritize energy independence and security, prompted by the political and economic consequences of the Arab oil embargo of 1973 (Giraud, 1983). In less than a couple of decades, electricity generation in France shifted from about 75 percent dependence on oil to about 75 percent dependence on nuclear energy. Electricité de France, the main electricity generation and distribution company, owns and operates the country's 56 power reactors. Thus, the French nuclear power system has centralized control in one large utility company.

The low generation costs of this nuclear fleet may be due in large part to an early commitment to standardization on just a few LWR designs (900 MWe, 1300 MWe, and 1450 MWe, all pressurized water reactors). As a result, the average cost of electricity in France is significantly lower than that in the European Union as a whole; for example, in neighboring Spain and Germany, prices are 46 percent and 79 percent higher, respectively, than in France (Selectra, 2021). Furthermore, France is the world's largest net exporter of electricity (WNA, 2022c). Standardization, however, could have the disadvantage that an equipment failure in one reactor could result in the shutdown of all reactors in that design class, thereby risking shutting down a large fraction of the nuclear power fleet (Ramana and Saikawa, 2011). In 2016, the French fleet manifested this type of flaw when about 20 reactors were shut down to address a carbon segregation problem in the steam generators' lower plates (Les Échos, 2016).

In 2015, the French Nuclear Assembly passed the Energy Transition for Green Growth law, which endorsed France's long-standing policy on energy security but sought to increase the share of renewable energies, such as solar and wind, and decrease the share of nuclear energy to 50 percent by 2025. In part, this law recognized that many of the reactors are nearing their nominal 40-year end of operational life. In November 2017, Environment Minister Nicolas Hulot noted that this goal was unrealistic, postponing the reduction to 2030 or 2035 and seeking to extend the operational life of the reactors. In November 2021, French President Emmanuel Macron announced plans to begin construction of new nuclear reactors in order to maintain the share of nuclear power at 50 percent of the electricity mix. In January 2022, the minister for ecological transition stated a target date of 2035–2037 for the new reactors to be commissioned (WNA, 2022c).

The relative stability of French energy policy results from general agreement within the French government on this issue and the absence of major policy shifts in the administration after each election, even considering the implications of the 2015 Energy Transition for Green Growth law. The size and integration of its nuclear industry, owned and controlled in large part by the French government, has allowed for a significant influence on the global nuclear industry and has had technology export benefits. Because nuclear power generates the vast majority of French electricity, France emits much lower amounts of carbon dioxide than other industrialized nations. Thus, nuclear power contributes significantly to French industrial, energy, and environmental policies.

Past experience and current policy on developing nuclear fuel cycles for multiple recycling of fissionable materials: Ensuring sustainable supplies of uranium has always been a top priority of French nuclear fuel cycle strategy. France has little domestic natural uranium. In addition, as mentioned earlier in this chapter, during the early decades of nuclear power, global resources of natural uranium were perceived as scarce, a view that drove France to devote R&D to fast reactors and reprocessing capabilities. However, development of fast reactor technology fell behind the deployment of LWR reprocessing technology. The French experience with fast reactors involved four main projects:

- Rapsodie (40 MWth): 1967–1983
- Phénix (250 MWe): 1973–2003
- Superphénix (1240 MWe): 1985–1997
- Phénix (170 MWe, after safety reevaluation): 2003–2009

According to the Commissariat à l'énergie atomique (CEA), these projects demonstrated the "excellent use of the uranium resource as well as the capability of these reactors to recycle the plutonium without any limitation in the number of recycling operations." At the same time, Phénix and Superphénix also showed how several material

selections could be unsuitable, and they shed light on sodium fast reactor safety functions, fuel handling, sodium leaks, operations, maintenance, and dismantling (Patel, 2019).

Given the lagging development of breeder reactors, the French program changed its near-term emphasis from the deployment of a closed fuel cycle with multirecycling to reliance on a fuel cycle in which recovered plutonium is recycled once in the form of LWR-MOX fuel (in the French 900-MWe LWR fleet) and the resulting spent LWR-MOX fuel is held in interim storage. This approach, however, has proven unable to prevent the accumulation of a large and growing stockpile of plutonium, totaling nearly 80 MT as of the end of 2020, in the form of both plutonium oxide and unrecycled MOX scrap (IAEA, 2021f). The growing inventory of plutonium-rich scrap generated by production problems at the MELOX MOX fuel fabrication plant has led to an urgent need to increase storage capacity for this material at La Hague (Johnstone, 2022). Faced with limited options for addressing this stockpile, France's long-term goal remains committed to closing the fuel cycle by recycling the plutonium in a fleet of cost-competitive fast reactors.

To progress toward this long-term goal, in 2006 the French government announced that the CEA would build a fourth-generation (Gen IV) 600-MWth fast reactor prototype, referred to as "ASTRID,"[30] by 2020. Initial connection of a first commercial-scale fast reactor to the French electrical grid was expected in the 2040–2070 time frame. However, in August 2019, after completing a detailed design, the plan for construction of ASTRID was shelved until at least the second half of the 21st century because of the sustained abundance and availability of uranium ores at low prices. In the short term, to stabilize the recovered plutonium stockpile and spent LWR-MOX fuel inventories created by the present policy, the CEA mission now includes an assessment, to be completed by around 2040, of the feasibility of multirecycling plutonium in the existing fleet of LWRs. Advanced European Pressurized Water Reactor (EPR) fuel designs are being investigated to enable plutonium multirecycling and stabilization of all spent fuel inventories. Two fuel assembly design concepts are being studied: (1) CORAIL, where the fuel assembly design contains both low-enriched uranium and MOX rods, and (2) MIX (also called MOXEUS), where the fuel assembly design consists of fuel rods containing a mixture of plutonium and enriched uranium oxides. Neutronic simulations indicate that introducing MIX and CORAIL in EPRs by the middle of the century could lead to a fast stabilization of spent MOX fuel and plutonium inventories (Martin et al., 2018). In addition, use of MOX assemblies, presently limited to the 900-MWe reactors, is scheduled to be extended to the 1,300-MWe reactors by the end of the decade.

Geologic repository: ANDRA, the French National Agency for Waste Management, has responsibility for nuclear waste management. In 1991, the "Bataille"[31] law was enacted, which defined three areas of waste management research: (1) separation and transmutation[32] of long-lived radioactive materials in the spent fuel (including minor actinides and some fission products); (2) final disposal of the waste in a geologic repository; and (3) interim storage for up to 300 years. In addition, the Bataille law required a 15-year period after which (in 2006) the French Parliament would create follow-on legislation to specify the path for the French nuclear waste management program.

The 2006 enactment of clarifying legislation included the following:

- Directed ANDRA to receive approval for and operate a permanent geologic[33] repository by 2015[34] and 2025,[35] respectively.
- Delayed a decision on the implementation of transmutation, possibly until at least 2040, corresponding to a possible time frame for building a new reprocessing facility with updated technology to replace La Hague. For the new facility, three main reprocessing R&D lines were to be maintained, not necessarily

[30] ASTRID = Advanced Sodium Technological Reactor for Industrial Demonstration.

[31] The law was named after Christian Bataille, a member of the French National Assembly.

[32] "Transmutation for waste management purposes is to convert a long-lived radionuclide that is potentially troublesome at a waste disposal site to a shorter-lived or stable nuclide by exposing the troublesome nuclide to a high flux of neutrons for a sustained time" (National Research Council, 1996).

[33] From 1991 to 2006, ANDRA opened an underground research laboratory in a clay formation in Bure (Meuse), France.

[34] Approval from the French government of the Bure site (CIGEO project, Centre Industriel de Stockage GEOlogique) was obtained in 2013.

[35] In 2021, ANDRA was preparing the license application. If authorization is granted, industrial disposal operations will start around 2035.

separate from each other: evolution of the PUREX process to coextract uranium and plutonium and possibly neptunium (COEX™ process); selective separation of minor actinides (DIAMEX-SANEX process); and group extraction of all actinides for homogeneous recycling in fast reactors (GANEX). (COEX, DIAMEX-SANEX, and GANEX are described in technical detail in Chapter 4.)

- Did not take any action on long-term interim storage because of the lack of experience of the French industry on this topic.

Bottom line: In the mid-1970s, France instituted a planned energy policy to improve energy independence and security by building a fleet of pressurized water reactors (PWRs). Reliance on nuclear generation for electricity supply reached as high as 80 percent. As of 2022, it stands at about 70 percent and is expected to stabilize at about 50 percent by 2035. The current fleet of PWRs and any new third-generation PWRs—EPRs—are envisioned to continue to dominate the French nuclear fleet through most of the 21st century. According to an Orano representative who presented to the committee in September 2021, the short-term objective (by the end of the 2020s) is to demonstrate mixed oxide fuel in the 1,300-MWe PWRs. The midterm (in the 2040s) ambition is deployment of next-generation mixed oxide fuels to support multirecycling PWRs. In the second half of the 21st century, a fast reactor prototype might be started, with the potential for a future power reactor fleet composed almost exclusively of fast reactors using a closed fuel cycle (Gay, 2021). This scenario, if implemented, would result in France being nearly energy independent for its electric power generation, assuming that the renewable sources also help substitute for use of fossil fuels for electricity production.

2.6 INSIGHTS ABOUT MERITS AND VIABILITY OF FUEL CYCLE OPTIONS FOR EXISTING LWR TECHNOLOGIES

The two partial fuel cycles referred to as once-through and monorecycle were not anticipated to be options during the early days of nuclear power development and deployment. Spent LWR fuel was intended to be reprocessed rather than disposed of in a geologic formation, and recovered plutonium was intended to be recycled in fast reactors, rather than in LWRs, given the much better nuclear properties of plutonium in a fast neutron environment (i.e., in fast reactors) compared with those in a thermal neutron environment (i.e., in LWRs). Both once-through and monorecycling systems are the direct result of adjustments required in light of the lack of progress in advancing fast reactors commercially.

The once-through option is likely to remain an "open" option, meaning that the spent fuel will be stored eventually in a geologic repository and is unlikely to enter a more advanced fuel cycle. The monorecycle option, on the other hand, may end up as either an open fuel cycle or become part of an advanced fuel cycle, as illustrated in Figure 2.3. Many countries (e.g., Belgium, Germany, Switzerland) that originally had reprocessing contracts have generally opted for monorecycling rather than storage but have no intent to further reprocess the spent mixed oxide fuel, which is now earmarked for final disposal. In this case, the monorecycle option will end up being an open fuel cycle. For France, the spent mixed oxide fuel may enter an advanced fuel cycle if and when fast reactors become commercially or strategically viable.

The front end of the once-through fuel cycle for thermal LWRs is well proven in dozens of countries, providing low-enriched uranium fuel (<5 percent uranium-235) at relatively low fueling costs. Globally, considering the current demand for nuclear power, uranium supplies are abundant and plentiful enough based on known resources at current uranium prices to ensure supply of these reactors for at least the next century. The series of steps to fuel an LWR the first time are the same whether or not the fuel cycle includes reprocessing. After reprocessing, a dedicated mixed oxide fuel fabrication facility is needed, which—because of the presence of plutonium—must meet safety and security requirements additional to those required for uranium oxide fuel fabrication. If the reprocessed uranium is recycled, the conversion and enrichment facilities will have to be appropriately modified to accommodate the added beta-gamma radiation associated with reprocessed uranium, as well as to add a separate, dedicated centrifuge cascade if this path is chosen to reenrich reprocessed uranium.

In contrast, the back end of the once-through fuel cycle is not complete. As Chapter 5 makes clear, all nuclear fuel cycles, open or closed, will require geologic repositories. The lack of geologic repositories for permanent

disposal of the spent nuclear fuel and high-level waste presently constitutes the biggest challenge for the national programs that have adopted an open fuel cycle approach. As a result, by both default and necessity, the nuclear industry has figured out how to temporarily manage waste safely in interim storage. In particular, utilities have had to increase the capacity of cooling pools for storing spent fuel assemblies at reactor sites and transfer an increasing fraction of their spent fuel inventory into dry (casks) or wet (pools) interim storage systems. In the United States, dry storage has been widely implemented at the reactor sites. While this system is working, it is not considered a permanent solution. Current practices are optimized for at-reactor-site storage, but not necessarily for transportation and disposal (Freeze et al., 2021), and there may be a need to repackage all of U.S. existing spent fuel prior to disposal in a geologic repository. Similarly, storage approaches are being used for the spent mixed oxides, high-level waste, and other products created by monorecycling.

A merit of the once-through cycle is its proliferation and theft resistance. Chapter 6 examines nonproliferation and security risks for advanced reactors and their associated fuel cycles. Low-enriched uranium at the 3–5 percent enrichment level is not weapons-usable material. Moreover, by not reprocessing and thus not separating out plutonium from highly radioactive fission products, the once-through cycle provides further protection against theft or unauthorized access to the plutonium; this is known as the "spent fuel standard" (DOE, 1999).

As was previously discussed, the original motivation for an advanced fuel cycle relying on the recovery of plutonium from the LWR spent fuel and recycling of the plutonium in a fast reactor was to extend fissile resources. As concerns for the adequate supply of natural uranium resources progressively subsided, the interest in such an advanced fuel cycle shifted to its potential merit to waste disposal, and more specifically to separation and transmutation of minor actinides. These topics are reviewed in more detail in Chapter 4.

3

Potential Merits and Viability of Advanced Nuclear Reactors and Associated Fuel Cycles

This chapter reviews the current U.S. government and Generation IV International Forum programs for the development of advanced nuclear reactors and associated fuel cycles in response to the first charge of the statement of task, which calls for an evaluation of the merits and an assessment of the viability of different nuclear fuel cycles, including fuel cycles that may use reprocessing, for both existing and advanced reactor technology options. The evaluation and assessment of existing options for operating commercial light water reactors (LWRs) were described in Chapter 2.

In this chapter, the committee provides its summary, findings, and recommendations up front (Section 3.1) and then describes international cooperative activities and the different types of advanced reactors (Section 3.2), programs to support their development (Section 3.3), and available and needed infrastructure for prototyping and testing these developing technologies (Section 3.4). Given the task to the committee to consider technologies that could be deployed by 2050, the committee provides perspectives, based on its information gathering and collective knowledge and judgment, on the status of technological development for each advanced reactor type. (Chapter 4 provides more details on the development and infrastructure needed to support fuel cycles for advanced reactors.) The committee notes that an assessment of the cost competitiveness of advanced reactor systems as compared with other energy sources is beyond the scope of this study.

3.1 CHAPTER 3 SUMMARY, FINDINGS, AND RECOMMENDATIONS

The Generation IV International Forum categorizes *advanced reactor systems* as including the following: (1) very-high-temperature reactor, (2) gas-cooled fast reactor, (3) sodium-cooled fast reactor, (4) lead-cooled fast reactor, (5) molten salt reactor, and (6) supercritical water-cooled reactor (SCWR). The integral pressurized water reactor, a small power reactor that leverages the technological infrastructure of the existing large-power pressurized water reactors, is also considered an advanced design according to the definition given by the Nuclear Energy Innovation and Modernization Act of 2019 (NEIMA) (Public Law 115-439). The U.S. government currently supports research and development (R&D) for all of these systems except the SCWR because of their commitments to support only those systems relevant to domestic R&D funded by the U.S. Department of Energy's Office of Nuclear Energy (DOE-NE) or with U.S.-based industry interest. Most of the advanced reactors being developed in the United States are *small modular reactors*, defined as having a notional power output of less than 300 MWe (megawatts electric) and envisioned for factory construction and modular installation. Almost all developers told

the committee that they are planning on an open, once-through fuel cycle at least for the near to intermediate terms, although several developers noted that in the longer term, their technologies have the potential of recycling spent nuclear fuel.

Considerable additional work and time are required before any of the non-LWR concepts could reach commercial deployment, and this chapter outlines the challenges faced by reactor developers. DOE-NE has programs that support the development of the advanced reactor and associated fuel cycles, but it will have to make difficult decisions in the coming years based on budgetary constraints. Finally, while the advanced reactors under development have several potential merits, these have yet to be demonstrated, and there are no operating prototypes in the United States for any of these reactors.

Finding 3: Government support to help bring advanced reactor technologies to commercial deployment will take substantial financial and technical resources. Specifically, budget limitations will require the U.S. Department of Energy (DOE) to make difficult decisions about its advanced reactor research and development programs to guarantee support, via industry cost sharing, for a *few promising* advanced reactor technologies and associated fuel cycle infrastructure in the next several years. If the Advanced Reactor Demonstration Program is funded consistently and fully by both the government and private industry through completion, information such as costs, reliability, project management, and manufacturing feasibility gained from this program will be key to helping DOE in its decision-making process.

Recommendation A: Using data from the Advanced Reactor Demonstration Program and the U.S. Department of Energy's (DOE's) research and development programs over the next several years, DOE should select and support, with industry cost sharing, the development of a *few promising* advanced reactor technologies and fuel cycles that can be potentially deployed by 2050 and achieve goals described in the Nuclear Energy Innovation Capabilities Act of 2017 (NEICA). DOE should develop a clear and transparent decision-making process based on criteria and metrics that can guide its programs and associated budget decisions going forward. With NEICA's goals as guidance, DOE's criteria should include (1) science-based estimates for improved fuel utilization and reduced waste yields compared with the existing light water reactor (LWR) fleet; (2) the development of acceptable waste forms and disposal options; (3) the implementation of enhanced safety throughout the entire fuel cycle, similar to that demanded for reactor design and operation; and (4) a level of proliferation resistance comparable to the LWR once-through cycle. DOE should also factor into its decision-making process the effort required and cost estimates for establishing advanced fuel cycles, including the manufacturing base and supply chain infrastructure required to support them. However, industry will have the primary responsibility for reactors that can be commercially deployed in the U.S. market.

Finding 4: Most of the advanced reactors, especially the non–light water reactors, will confront significant challenges in meeting commercial deployment by 2050. While at least 10 advanced reactor developers currently aim to deploy their technologies by 2050 in the United States,[1] there are no currently operating fueled prototypes of any of these specific advanced reactor designs in the United States; there are, however, some demonstration and commercial units of similar reactor designs in operation internationally. Moreover, the vast majority of advanced reactors are still in the early design phase. Depending on the maturity of the technology, advanced reactor developers face a range of challenges to bringing the proposed technologies to commercialization, including little or no direct operational experience of some designs at engineering scale; the lack of adequate capabilities to develop, test, and qualify advanced fuels and materials; and as a result, the potential considerable time for regulatory approval.

Recommendation B: To support the development and deployment of advanced reactor technologies, Congress and the U.S. Department of Energy (DOE) need to provide or ensure access to materials testing and fuel qualification capabilities essential to advancing these technologies. Accomplishing this

[1] As of January 2022, this number of developers had submitted applications or preapplications to the U.S. Nuclear Regulatory Commission.

requires a coordinated plan involving DOE's Office of Nuclear Energy, Office of Science, and domestic and international user communities. The plan should consider a full range of alternatives in meeting both short- and long-term needs.

Finding 5: Of the advanced nuclear reactor technologies currently in development, small modular reactors based on light water reactor (LWR) technologies are furthest along toward being connected to the electrical grid. This is because they can leverage the existing LWR and fuel cycle infrastructure and because these technologies have received government and private-investor financial support for more than a decade.

Finding 6: The common perception that the thorium-232/uranium-233 fuel cycle will generate less plutonium and minor actinides (therefore reducing the radioactive hazard of its spent fuel compared with that from the uranium-235/plutonium fuel cycle) is incorrect. Overall, because of the decay of associated actinide products, thorium-based fuels have short- and long-term radiotoxicities (hazards) comparable to uranium-based fuels.

3.2 TYPES OF ADVANCED REACTORS AND ASSOCIATED FUEL CYCLES

A large number of advanced reactors are being designed that differ significantly from the current fleet of thermal LWRs deployed in the United States. Like LWRs, the advanced reactors will mostly use the fission of uranium-235 in nuclear fuel. Unlike LWRs, many advanced reactors will rely on non-water-based coolants, such as gases (e.g., helium, carbon dioxide), liquid metals (e.g., sodium, lead), and molten salts (e.g., fluoride-, chloride-based salts). Some of the advanced concepts depend on first transforming the fertile isotope thorium-232 to the fissile uranium-233, and then extracting energy from the fission of uranium-233. Advanced reactors will sustain the fission reaction with either neutrons moderated to thermal energies (less than 1 eV [electron volt])[2] or unmoderated systems that use fast neutrons (energies from ~0.1 to 1.0 MeV [mega electron volt]). Thermal reactors need moderators to slow down neutrons to thermal energies, and usually light water, heavy water, or graphite blocks are used as moderators.

Fast reactors were originally designed to use plutonium-239 as their primary fuel because, compared with uranium-235, plutonium-239 has a smaller capture-to-fission ratio for fast neutrons, produces more neutrons per fission regardless of the neutron energy, and generates more neutrons per neutron absorbed. Although the United States has a supply of plutonium-239 declared excess to its nuclear weapons program, it has no plans to use this supply for fast reactors and will most likely not reprocess spent nuclear fuel to produce plutonium for fast reactors for the foreseeable future. Thus, fast reactors will have to rely on uranium-235 for their fuel, except for those that could use uranium-233 produced from a thorium fuel cycle.

To sustain the chain reaction, fast reactors will require higher enrichments of uranium-235 to counteract both the much lower probability of fission of uranium-235 induced by fast neutrons and the probability of fast neutron capture by uranium-238. As a result, these reactors will need at minimum high-assay low-enriched uranium (HALEU) with uranium-235 enrichment greater than 10 percent but less than 20 percent. (See Chapter 4 for a discussion of HALEU production infrastructure and supply.) Fast reactors can operate as burners, converters, or breeders based on the conversion ratio (CR), or ratio of fissile material produced to material consumed. Burner reactors consume more fissile material than they produce (CR <1); converter reactors produce about the same amount of fissile material as they consume (CR ≈1); and breeder reactors produce more fissile material than they consume (CR >1).

Fuel options for advanced reactors include solid fuel composed of uranium, thorium, or uranium/plutonium in physical forms, such as refractory ceramics (predominantly oxides or mixed oxides and nitrides, carbides, or silicides) and metals (typically metal alloys), or as liquid fuel (fluoride or chloride salts of uranium, thorium, and plutonium) used in molten salt reactors. Solid fuel can have different shapes, such as fuel rods, pebbles, and cylindrical fuel compacts. A fuel rod consists of fuel pellets enclosed by protective cladding. Instead of fuel pellets, the fuel can be

[2] The ideal Maxwell–Boltzmann distribution of thermal neutron energies has its mode, or most probable energy, at 0.025 eV, which corresponds to a temperature of 20°C, or typical room temperature.

in the form of coated fuel particles (e.g., TRistructural ISOtropic [TRISO] particles). Coated fuel particles can be arranged into cylindrical fuel compacts that are inserted into prismatic graphite blocks, forming fuel elements (e.g., prismatic graphite block, as in Framatome's Steam Cycle High Temperature Gas-Cooled Reactor [SC-HTGR]) or pebbles (as in X-energy's Xe-100 pebble-bed reactor). (Chapter 4 provides details on the advanced reactors' fuels and fuel fabrication.) Coolants are required to remove heat from the reactor core; common coolant materials are water, gas (helium, carbon dioxide, nitrogen), liquid metal (lead, sodium), and liquid salt (FLiBe [2^7LiF-BeF_2]). Fuel assemblies and fuel rods are usually arranged in a square or hexagonal lattice, or randomly, as in pebble-bed reactors.

3.2.1 Generation IV International Forum's Advanced Reactor Systems

In 2000, the U.S. government proposed the formation of the Generation IV International Forum (GIF) as a mechanism to promote international cooperation in the R&D of advanced nuclear energy systems. As of 2022, the GIF member countries are Argentina, Australia, Brazil, Canada, China, France, Japan, the Republic of Korea, Russia, South Africa, Switzerland, the United Kingdom, the United States, and Euratom (representing 27 European Union member countries).

GIF provides a framework for categorizing the major classes of advanced reactor systems. The Generation IV (Gen IV) framework was defined from 2001 to 2002 via identifying promising reactor technologies to examine, establishing technology goals, and setting a legal framework for cooperation. An expert group considered more than 100 designs and then selected the six most promising systems for further R&D. These design systems are (1) the gas-cooled fast reactor, (2) the lead-cooled fast reactor, (3) the molten salt reactor, (4) the sodium-cooled fast reactor, (5) the supercritical water-cooled reactor (SCWR),[3] and (6) the very-high-temperature reactor (VHTR)[4] (see Figure 3.1).

The goals for the Gen IV systems are

- Sustainability for the long-term fuel supply and for minimization of waste and the long-term burden of this waste;
- Safety and reliability to achieve a very low likelihood and degree of reactor core damage and to eliminate the need for off-site emergency response;
- Economics that have life cycle cost advantage as compared to non-nuclear energy systems and financial risk comparable to other energy systems; and
- Proliferation resistance and physical protection to have unattractive materials diversion pathways and enhanced physical protection against terrorist attacks (Kamide, 2021).

The U.S. government supports R&D for five of the six Gen IV systems; it does not support R&D for SCWR because DOE-NE has committed to support only those systems that are relevant to DOE-NE's funded domestic R&D programs or have U.S.-based industry interest (Caponiti, 2022). Progress related to the development of the six reactor systems along with the cross-cutting R&D activities under the purview of GIF task forces and working groups can be found in the Gen IV International Forum Annual Reports (GIF, 2022b).

DOE's framework for advanced reactors designs is similar to that of GIF, but DOE more highly prioritizes versatility compared with earlier reactor generations, especially the ability to provide nonelectrical services, such as desalination, process heat, and hydrogen production, as an additional goal of the advanced reactor designs. Like the Gen IV systems, DOE's advanced designs have potential improvements that could manifest themselves in a number of ways, such as inherent or passive safety features, simplified or modular designs for ease of fabrication,

[3] SCWR is a reactor concept for operating at supercritical pressures with light water as coolant and moderator. Supercritical water has indistinguishable densities for liquid and steam, and thus, SCWRs can eliminate the need for pressurizers and steam generators (as are used in pressurized water reactors [PWRs]). Consequently, an SCWR is simplified compared with a PWR and can operate at higher thermal efficiencies—approximately 45 percent as compared with 33 percent for a PWR. The SCWR concept is not considered in this report.

[4] According to GIF, "The VHTR is a next step in the evolutionary development of high-temperature gas-cooled reactors. It is a graphite-moderated, helium-cooled reactor with thermal neutron spectrum. It can supply nuclear heat and electricity over a range of core outlet temperatures between 700 and 950°C, or more than 1,000°C in future" (GIF, 2022a).

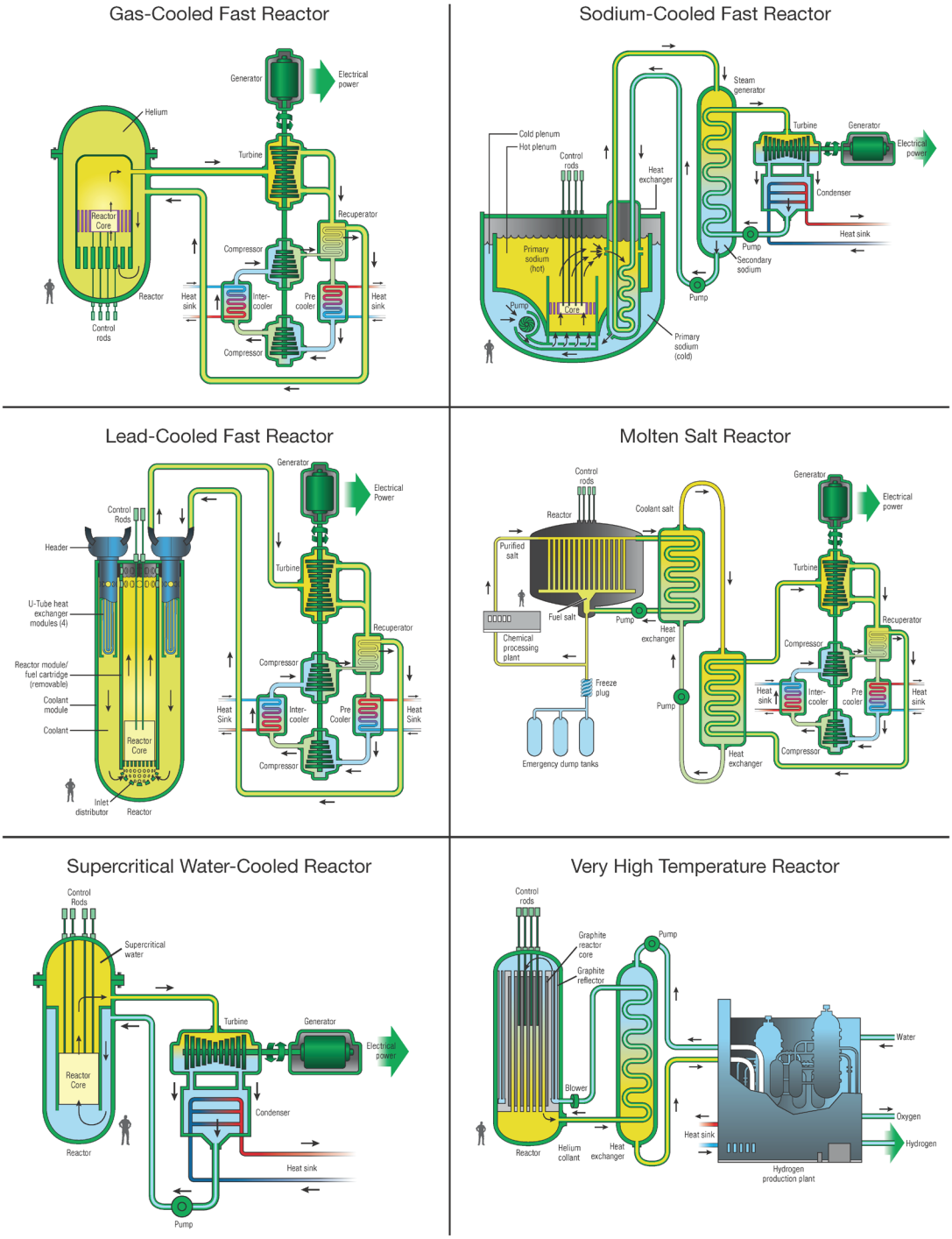


FIGURE 3.1 Schematics of the six advanced reactor concepts in the Generation IV International Forum. SOURCE: GIF and NERAC (2002).

scalability and enhanced load-following capabilities to complement sources of renewable energy, increased safety of accident-tolerant materials, and fast neutron spectrums for increased fuel utilization via closed fuel cycles (Arostegui and Holt, 2019). However, it should be noted that the Gen IV designation refers strictly to the six designs meeting the stated goals of GIF, and some DOE-supported advanced reactor designs do not fall under the Gen IV classifications. In DOE's broader definition, advanced reactors are further subdivided into categories of microreactors, small modular reactors, and full-sized reactors, described as follows:

- Microreactors are factory fabricated and fully assembled; transportable by truck, ship, railcar, or air to residential, remote commercial, or military locations; installed and operational quickly; and self-adjusting with passive safety systems and small support staff (DOE-NE, 2021b).
- Small modular reactors are factory fabricated; can transport major components from factory fabrication locations to the plant site by rail or truck; and can include designs that provide safeguards, security, and nonproliferation advantages. Both advanced water-cooled and non-water-cooled reactors (gas, liquid metal, and molten salts) can be configured as small modular reactors (DOE-NE, 2021c).

Most of the advanced reactors being developed in the United States are small modular reactors (SMRs). In addition to the potential benefits described by DOE, motivations for SMR development include smaller plant footprint and potentially lower operation and maintenance costs, and lower up-front capital costs (excluding the to-be-expected higher costs of the first-of-a-kind unit), as well as improved safety. However, because SMRs have yet to be commercially deployed in the United States, these systems have to demonstrate their operational economic competitiveness compared with larger nuclear power plants and other nonnuclear energy systems.[5] The non–light water SMRs will also need to demonstrate that they can meet licensing requirements that include safety assessments by the U.S. Nuclear Regulatory Commission (U.S. NRC).

All the Gen IV systems except for the SCWR, which is not supported by DOE, are described in Section 3.2.3. Also included in Section 3.2.3 is a relatively detailed description of NuScale's light water–cooled SMR, which represents one of the most technologically mature light water SMRs currently under development.

3.2.2 International Collaboration and Partnerships

In addition to supporting the development of almost all of the Gen IV systems, DOE has cooperative agreements with other international partners through its Office of International Nuclear Energy Policy and Cooperation (INEPC). Relevant to this study, INEPC has worked with countries such as France, Japan, and Russia investigating advanced nuclear fuels. As discussed in Section 3.5, cooperative activities involve use of materials testing reactors and other facilities for evaluation of advanced reactors' fuels and materials. To facilitate cooperation, INEPC can use bilateral technical collaboration arrangements, memorandums of understanding, technical action plans, the International Nuclear Energy Research Initiative (I-NERI), and the International Nuclear Cooperation framework. Pertinent to this study, in 2001 DOE-NE established I-NERI to perform R&D with international partners at facilities developing advanced nuclear energy systems and to support R&D linked to DOE-NE's principal research programs. I-NERI collaborators include Canada, the European Union, and the Republic of Korea (DOE-NE, n.d.-a).

In 2015, the Nuclear Energy Agency (NEA) of the Organisation for Economic Co-operation and Development (OECD) began a wide-ranging initiative on nuclear energy innovation, called the Nuclear Innovation 2050 project, with emphasis on fostering international cooperation, especially among OECD nations. The scope of this project's technology areas includes reactor systems design and operation, fuels and fuel cycles, waste management and decommissioning, and nonelectrical applications of nuclear power. Cooperative activities have involved U.S. national laboratories, such as Argonne National Laboratory and Idaho National Laboratory partnering with Belgium's SCKCEN, France's CEA, and Japan's Atomic Energy Agency

[5] This topic is covered in the parallel National Academies study on advanced nuclear reactors.

on a variety of topics, including the following of relevance to this study: improving the fuels qualification[6] process to accelerate industrial deployment, advanced structural materials for Gen IV systems, advanced fuel cycle chemistry, and advanced components for Gen IV to foster safety and economics.[7] The cooperative activities described above could help optimize financial resources for each country involved (Todd, 2021).

3.2.3 Advanced Reactor Types

The committee received presentations from representatives of all the advanced reactor developers listed in Table 3.1; in total, the committee heard from 15 advanced reactor developers on 17 different reactor designs (see listings of presentations in Appendix B). Detailed technical design considerations, safety aspects, electricity generation, and nonelectricity applications of these advanced reactors are addressed in the parallel National Academies study Laying the Foundation for New and Advanced Nuclear Reactors.

As an indicator of the number of advanced reactor designs that have made steps toward potential deployment in the United States by 2050, 10 developers have submitted for 11 designs license applications or preapplications to the U.S. NRC. These are

- "NuScale—light-water SMR [small modular reactor]
- Oklo Aurora—fast micro-reactor[8]
- GEH BWRX-300—light-water SMR
- General Atomics EM2—gas cooled fast reactor
- Holtec SMR-160—light-water SMR
- Kairos Power—molten salt-cooled reactor with TRISO fuel
- Terrestrial Energy IMSR—thermal molten salt reactor
- TerraPower Natrium—sodium-cooled fast reactor
- TerraPower MCFR—fast molten salt reactor
- Westinghouse eVinci—micro-reactor
- X-energy Xe-100—high-temperature gas reactor" (Nichol, 2021).

3.2.3.1 High-Temperature Gas-Cooled Reactor

High-temperature gas-cooled reactors (HTGRs) are graphite-moderated, helium-cooled thermal reactors that use TRistructural ISOtropic (TRISO) particle fuel and have high outlet temperatures of 700–1,000°C, which leads to higher thermal efficiencies (Arostegui and Holt, 2019). TRISO fuel consists of uranium oxycarbide or uranium dioxide (up to 20 percent uranium-235 enrichment) encased in carbon and silicon carbide; these fuel particles (1 mm in diameter) are stable up to 1,600°C (WNA, 2020b). Two primary HTGR designs have been proposed: pebble-bed and prismatic block, named for the different shape of the fuel elements used (MIT, 2018). Pebble-bed designs contain hundreds of thousands of spherical fuel elements called "pebbles" (up to 6 cm in diameter), which are cycled through the reactor core; each pebble contains thousands of TRISO fuel particles. Prismatic block designs use hexagonal graphite blocks in the reactor core, with holes in graphite blocks for insertion of cylindrical TRISO fuel compacts and holes for circulation of helium coolant.

The primary fuel cycle research needs for HTGR reactors relate to the fabrication, qualification, and disposal of TRISO fuel, as discussed in Chapters 4 and 5; the management of radioactive graphite dust, as discussed in

[6] While there is no explicit definition or use of fuel qualification in U.S. NRC rules or regulations, the report *Fuel Qualification for Advanced Reactors* (NUREG-2246 [U.S. NRC, 2022a]) defines the term as "the overall process (planning, testing, analysis, etc.) used to obtain qualified fuel," where *qualified fuel* is defined as "fuel for which reasonable assurance exists that the fuel, fabricated in accordance with its specification, will perform as described in the safety analysis."

[7] To access papers on these topics, see https://www.oecd-nea.org/jcms/pl_21829/nuclear-innovation-2050-ni2050.

[8] In January 2022, the U.S. NRC denied the Oklo Aurora combined license application because of the company's failure to supply sufficient information, despite repeated requests by agency staff (U.S. NRC, 2022d). Soon after the denial, Oklo announced that it will respond and resubmit its application.

TABLE 3.1 Classification of Advanced Reactor Systems by Coolant, Neutron Spectrum, Fuel Type, Uranium Enrichment, and Fuel Cycle Options

	Neutron Spectrum					Fuel Cycles	
Coolant	Thermal	Fast	Company Design Examples	Fuel/Cladding/ Lattice	Uranium Enrichment	Primary[a]	Options[b]
Water	Small modular light water reactor		NuScale Power Module (NPM)	UO_2/Zr-based Zircaloy-4 or advanced cladding/ square	<5% LEU	Once-through cycle (OTC)	U and Pu recycle
Helium	High-temperature gas-cooled reactor		X-energy (Xe-100)	UCO/TRISO/ pebble-bed	HALEU (15.5%)	OTC—high burnup	
	Steam cycle high-temperature gas-cooled reactor		Framatome (SC-HTGR [Steam-Cooled HTGR])	UC/TRISO/ cylindrical compacts/ prismatic block	HALEU (14.5%)	OTC—high burnup	Other fissile/fertile nuclides
	Microreactor		BWXT (BANR [BWXT's Advanced Nuclear Reactor])	UC/TRISO/ prismatic block	HALEU (19.75%)	OTC	
		Gas-cooled fast reactor	General Atomics (EM^2 [Energy Multiplier Module])	UC/SiC	HALEU (14%) & DU	OTC—convert and burn	Recycle
Liquid Metal		Sodium fast reactor	TerraPower (Natrium)	U/Steel	HALEU (10–18.5% core average)	OTC	Once-through breed-and burn; closed fuel cycle via multirecycling
			ARC-100	U-Zr/Steel	HALEU (10.9-15.5%)	OTC	Closed fuel cycle via multirecycling
		Lead Fast Reactor	LeadCold (SEALER-55)	UN/Steel	HALEU (12%)	OTC	Potential recycling of spent fuel
		Lead Fast Reactor	Westinghouse (LFR)	UO_2/Steel—phase I; UN/Steel or SiC—phase II	HALEU (13.8% UO_2) or (11.9% UN)	OTC—UO_2	Potential MOX for Pu recycle, if/when pursued; partially closed LWR-Pu or closed LFR-Pu

Molten Salt	Fluoride salt-cooled high-temperature reactor		Kairos (KP-X FHR)	UCO/TRISO-AGR/pebble-bed	HALEU (19.75%)	OTC	
	Molten fuel salt reactor—fluoride		Terrestrial Energy (IMSR-400 [Integrated Molten Salt Reactor])	U fluoride (liq.)	<5% LEU or HALEU	Modified OTC: 1/3 fuel salt OTC to storage and 2/3 recycled directly to new core unit	Multiple options
			ThorCon	Th/U fluorides (liq.)	HALEU (19.75%)	U recycling	Closed fuel cycle
			Flibe Energy (Liquid Fluoride Thorium Reactor) ^{233}U breeder	ThF_4 (liq.)/UF_4 (liq.)		Closed fuel cycle via continuous multirecycling	
Molten Salt		Molten fuel salt reactor—chloride	TerraPower (Molten Chloride Fast Reactor)	UCl_3 or $PuCl_3$ (liq.)	HALEU (12%) and U_{dep}	OTC—high burnup	Net breed and burn—multiple options
			MOLTEX (Stable Salt Reactor—Wasteburner)	$(Pu\text{-}U_{dep})Cl_3$ (liq.)/steel		Closed fuel cycle—pyroprocessing	
Heat—Pipe		Microreactor	Oklo (Aurora Powerhouse)	U-Zr/Steel	HALEU (12-19.75%)	Closed fuel cycle—pyroprocessing (electrorefining)	
	Microreactor		Westinghouse (eVinci)	UC/TRISO/prismatic block	HALEU (19.75%)	OTC	

[a] Primary fuel cycle as intended currently by the advanced reactor developers.

[b] Optional fuel cycle that could be developed depending on policy, regulatory, and economic drivers.

NOTE: HALEU = high-assay low-enriched uranium; HTGR = high-temperature gas-cooled reactor; LEU = low-enriched uranium; LFR = lead-cooled fast reactor; LWR = light water reactor; MOX = mixed oxide; TRISO = TRistructural ISOtropic; UC = uranium carbonide; UCO = uranium oxycarbide; U_{dep} = depleted uranium.

SOURCE: Adapted from MIT (2018).

Chapter 5; and the development of safeguards for material accountancy and control of TRISO pebbles, as discussed in Chapter 6. Examples of HTGRs under development are X-energy's Xe-100 (TRISO pebble-bed), Framatome's SC-HTGR (TRISO prismatic block), and BWXT's Advanced Nuclear Reactor (BANR) microreactor (TRISO prismatic block).

As to technology development, important lessons can be learned from experience in other countries in developing and deploying pebble-bed HTGRs. The German AVR, an experimental pebble-bed HTGR that operated from 1967 to 1988, experienced challenges with inadmissibly high core temperatures, contamination with metallic fission products and radioactive dust, and water ingress (Moorman, 2008). The operational challenges encountered with AVR prompted additional research into improving metallic fission product retention in fuel elements, modeling core temperature behavior, and improving dismantling practices to avoid radioactive dust contamination (Moorman, 2008). In 2000, China entered an agreement with South Africa for cooperation in developing this technology. Starting in 1994, South Africa had invested significant resources in pebble-bed technology and had made progress, such as building a prototype fuel fabrication facility, but the planned test reactor was not constructed. In 2010, South African government funding was cut because of lack of investor interest in commercializing the technology (WNN, 2010). China has taken more than 20 years to develop and deploy a demonstration HTGR, and its experience could provide a potential pathway for U.S. developers of this technology. Briefly, Chinese researchers first built the prototype reactor HTR-10, which has a 10-MWe power rating, to thoroughly test materials, components, and the various systems, as well as to train a cadre of engineers, technicians, and operators for all aspects of the reactor's systems. This extensive prototyping experience led to scaling up to the HTR-PM (high-temperature [gas-cooled] reactor—pebble-bed modular), a 200-MWe commercial demonstration reactor with a 750°C operating temperature, resulting in about a 40 percent thermal efficiency. Construction of HTR-PM began in 2012, and the reactor went critical on September 12, 2021; the relatively long lead time can be attributed to development of first-of-a-kind technologies (Adams, 2021). In the United States, as discussed in detail in Section 3.3.2, X-energy's Xe-100 has recently received more than $1 billion of support via DOE's Advanced Reactor Demonstration Program (ARDP) and has a timeline to deploy a demonstration unit by 2028. X-energy has invested in developing TRISO fuel fabrication, as discussed in Chapter 4.

3.2.3.2 Gas-Cooled Fast Reactor

Gas-cooled fast reactors (GFRs) are fast neutron spectrum reactors that use ceramic-clad uranium carbide or nitride fuel and helium gas as the coolant (GIF, 2021b; WNA, 2020b). These reactors are expected to operate with high outlet temperatures of around 850°C and are typically envisioned for use in a closed fuel cycle with full actinide recycling. In the GFR design, heat is transferred from the primary helium circuit into a secondary helium/nitrogen circuit, which then powers a gas turbine.

The GFR is the only GIF reactor design that does not have an operating predecessor from which an experience base can be developed, so its introduction will almost certainly be as an experimental technology demonstration reactor. Fuel cycle R&D needs include fabrication and qualification of the fuel and cladding materials and development of a reprocessing system capable of actinide recycling from uranium carbide or nitride fuels.[9] The AIROX recycle process, a cyclic dry pyrochemical oxidation/reduction process, is being pursued to eliminate the need for enrichment after first core and leave fission products as the only waste (Gougar, 2018). According to Majumdar et al. (1992), the AIROX process applied to spent LWR fuel would generate 0.29 MT (metric tons) of cladding and hardware and 0.1 MT of semivolatiles as GTCC waste per MT of initial heavy metal of spent fuel.[10] The heat load from AIROX fuel would be similar to high-burnup fuel; consequently, AIROX-reprocessed fuel would be hotter than normal once-through spent fuel (Majumdar et al., 1992). The AIROX process produces

[9] For uranium nitride fuel, the (n,p) reaction on nitrogen-14 produces copious amounts of carbon-14, which becomes a waste management issue unless the nitrogen is heavily enriched in nitrogen-15. Because of the enrichment costs, any recycle process must recover nitrogen-15 in high yield.

[10] The text was corrected after release of the prepublication version of the report to clarify that it referred to the AIROX process as applied to spent LWR fuel.

fewer, but hotter, spent fuel assemblies. This must be factored into the repository design. General Atomics' EM^2 and Euratom's Allegro are other examples of proposed GFR designs.

The path to commercial deployment of the GFR will need to traverse uncharted territory given its lack of an operational predecessor. General Atomics has performed extensive research, especially on carbide fuel development,[11] and has outlined to the International Atomic Energy Agency (IAEA) the following timeline toward potential deployment: in 2010, conceptual design and development started; in 2023, high-risk development completed; in 2024, prelicensing vendor design review started in the United States; in 2029, engineering design complete; in 2030, construction of a prototype started in the United States; in 2032, potential commercial operation (GA, 2019). In comparison, the European consortium developing the Allegro reactor is seeking to deploy the first prototype GFR but recognizes that many years of development are required. This consortium was formed in 2010 and needed 5 years of preliminary conceptual work to reach the beginning of the design conceptual phase in 2015. The design conceptual phase is expected to take up to 11 years, until 2026. At that point, a decision will have to be made about whether to continue the Allegro project (Bělovský, 2019). As mentioned above, a major technical challenge is qualification of the carbide and nitride fuels. Another significant challenge is that the plant's materials would have to withstand high levels of radiation damage (such as neutron embrittlement) coming from the long-lived cores that are being proposed (MIT, 2018).

3.2.3.3 Sodium-Cooled Fast Reactor

Sodium-cooled fast reactors (SFRs) are fast neutron spectrum, high-power density reactors that are cooled with liquid sodium and use either metallic uranium/plutonium or mixed oxide fuel (GIF, 2021a; WNA, 2020b). The use of liquid sodium as a coolant allows low-pressure operation and provides more effective heat transfer than water, but an SFR requires an air- and moisture-free environment because sodium metal reacts violently with air and water, and the opaqueness of the sodium coolant presents monitoring and inspection challenges (Flanagan et al., 2015; GIF, 2021a). Incidents related to the sodium coolant—leaks, corrosion, fires, reactivity with water—have been key challenges for previous and currently operating SFRs (IAEA, 2013c). There are two primary reactor designs: pool-type, in which the primary heat exchanger remains contained in the reactor vessel, and loop-type, in which the primary heat exchanger is outside the reactor vessel (Flanagan et al., 2015). Both designs incorporate an intermediate heat exchanger to separate the primary heat exchanger from the power generation system. A range of reactor sizes is being considered, from small modular-type reactors that generate 50–150 MWe to large loop-type reactors that generate 600–1,500 MWe (GIF, 2021a).

Several fuel cycle variations are being proposed to support SFRs in the long term that will require additional development in the areas of fuel fabrication and reprocessing. These include multirecycling of plutonium and possibly also minor actinides with either metallic or mixed oxide fuel, as discussed in Chapter 4. Depending on the choice of fuel, issues requiring attention could include fuel swelling, fuel/cladding chemical and mechanical interactions, and fuel/coolant compatibility (Hill, 2021). Electrometallurgical pyroprocessing for use with uranium/plutonium metallic fuel is envisioned, while mixed oxide reprocessing would likely involve advanced aqueous processes. If uranium nitride fuel is used in an SFR, the same carbon-14 generation and waste management problem as seen with GFR fuel occurs if the uranium nitride fuel is not first highly enriched in nitrogen-15 (see footnote in Section 3.2.3.2). The pyrophoric sodium coolant (and, for some reactor designs, metallic fuel) that uses sodium to bond the fuel thermally to the cladding presents unique waste management challenges, as detailed in Chapter 5. Example SFR designs under development include TerraPower/GE-Hitachi's Natrium and Advanced Reactor Concepts' ARC-100.

Experimental and commercial SFRs have been operated worldwide dating back to the 1950s. At the present time, Russia has two operating SFRs, BN-600 and BN-800, and China is building two demonstration reactors (CFR-600). According to the IAEA ARIS (Advanced Reactors Information System) database,[12] other SFR designs (experimental, demonstration, and commercial) are being pursued by Japan, Russia, India, the Republic of Korea, and the United States.

[11] For a list of scientific papers related to General Atomics' EM^2 research, see https://www.ga.com/nuclear-fission/scientific-papers.
[12] See https://aris.iaea.org.

As to future technology development, Natrium has received significant financial support—more than $1 billion—from the ARDP, as discussed in Section 3.3.2, and equal levels of funding from its private investors. In November 2021, DOE announced its intention to invest nearly $2 billion in the Natrium demonstration reactor, which is planned to be built near a to-be retired coal plant near Kemmerer, Wyoming. As of late 2021, TerraPower is working on an application to the U.S. NRC for a permit to start construction by 2024. Under the ARDP, TerraPower has committed to achieving licensing, construction, and operation by 2028. Although there have been several experimental and commercially operational SFRs since the 1950s, DOE has described Natrium as a first-of-a-kind project (DOE-NE, 2021d). TerraPower has claimed that Natrium's "new plant architecture minimizes cost and construction time" but told the committee that a challenge is that non-LWR licensing requirements have not yet been fully defined (Neider, 2021).

3.2.3.4 Lead-Cooled Fast Reactor

Lead-cooled fast reactors (LFRs) are fast spectrum reactors that typically use metallic uranium or uranium nitride fuel and liquid lead or lead-bismuth eutectic coolant (GIF, 2021c; WNA, 2020b). A variety of LFR designs have been proposed, with outlet temperatures ranging from 300 to 750°C, though the only operational experience has been in Russia's Alfa-class submarines from 1968 to 1995 (WNA, 2020b, 2021d). The liquid lead or lead-bismuth eutectic employed as coolants are chemically inert to air and water and have high boiling temperatures, which provide potential safety benefits relative to sodium metal coolant (GIF, 2021c; Wallenius, 2021). While the use of lead-based coolants allows for low-pressure operation, the reactor must be maintained at high temperatures to prevent the lead from solidifying (GIF, 2021c). At such high temperatures, lead becomes corrosive to steel, and further research is required to develop alternative materials with higher durability under the required operating conditions. In 2021, LeadCold announced development of an aluminum alloyed steel, Fe-10Cr-4Al-RE, which is claimed to be resistant to lead corrosion (LeadCold, 2021). Lead and lead-bismuth coolants also pose challenges related to chemical toxicity, and long-term leaching into the environment must be prevented (IAEA, 2019e). Because of their high toxicity, lead and other heavy metals are regulated in the United States by the Environmental Protection Agency under the Resource Conservation and Recovery Act, which could potentially present a hurdle for reactor designs proposing to use such lead-based coolants. Furthermore, for both LFRs and SFRs, the opacity of the coolant and high-temperature operations complicate monitoring and maintenance procedures (GIF, 2021c; Wallenius, 2021). Likewise, LFRs and SFRs can use depleted uranium or thorium fuel and burn actinides from LWR fuel. Full actinide recycle is expected at maximum reactor technology implementation, requiring the development of advanced aqueous or nonaqueous pyroprocessing of spent fuel. As mentioned above, the use of uranium nitride fuel that is not highly enriched in nitrogen-15 leads to major carbon-14 waste management issues (see footnote in Section 3.2.3.2). Examples of LFRs under development include LeadCold's SEALER-55, Westinghouse's Lead Fast Reactor, and Russia's BREST-OD-300 and SVBR-100.

As to technology development, LeadCold told the committee in February 2021 that KTH (the Swedish Royal Institute of Technology and a research partner) "has developed methods for manufacture of uranium nitride permitting tailor made manufacture of this fuel at industrial scale" with use of 99.5 percent enriched nitrogen-15, although it acknowledges that this fuel type is "difficult to manufacture using conventional methods." Moreover, an "industrial scale supplier of ^{15}N has been identified," and Studsvik in Sweden could be a potential fuel fabricator (Wallenius, 2021). LeadCold aims to have a demonstration reactor of its SEALER-55 design built by 2030 at Oskarshamn, Sweden (WNN, 2021b).

Westinghouse's LFR presenters told the committee in October 2021 that their conceptual design is nearing completion, and commercialization might occur in the 2030s, according to Westinghouse's "staged approach to development." This approach leverages a network of test rigs, located primarily in the United Kingdom, the United States, and Italy, which are used to also support demonstration of key systems, components, and phenomena of the Westinghouse LFR. The testing involves corrosion/erosion, fuel manufacture, plant safety and reliability, lead's effects on materials' mechanical properties, component testing (e.g., on the fuel bundle and primary heat exchanger), and instrumentation (e.g., under-lead viewing). The start-up core is planned to use HALEU uranium

dioxide with D9-type austenitic steel cladding; future performance-enhanced cores could use uranium nitride with advanced steel or silicon carbide cladding (Ferroni et al., 2021).

3.2.3.5 Molten Salt Reactors

Molten salt reactors (MSRs) have a wide variety of design options but limited operational experience. Two experimental MSRs have operated: the 1950s-era Aircraft Reactor Experiment and the 1960s-era Molten Salt Reactor Experiment, both in the United States, and both thermal MSRs. A fast MSR has never been operated. A primary classification consideration is whether an MSR uses molten salt as coolant only or as fuel and coolant. GIF defines *molten salt reactor* as the latter, in which the fuel is dissolved in the salt coolant (GIF, 2021d). However, DOE, among others, has sometimes referred to the design being developed by Kairos Power as an MSR, although that concept is a fluoride salt–cooled reactor with solid TRISO fuel and is properly referred to as a fluoride salt–cooled, high-temperature reactor (DOE-NE, 2020).

MSRs offer the potential for high fuel-resource conservation and high actinide consumption via burnup. Liquid fuel MSRs can operate with a thermal, epithermal, or fast neutron spectrum and can use a variety of uranium- and thorium-based fuels. The fuel form can be as solid particle fuel in graphite similar to the core of a high-temperature gas-cooled reactor (HTGR) or dissolved in the molten salt to form a liquid "fuel salt" such as in Flibe Energy's design. The latter has the advantage that no solid fuel fabrication is required, but because gaseous fission products separate from the liquid fuel during operations, a fully open fuel cycle is not technically possible.[13] Molten salt coolants can be either a fluoride- (e.g., FLiBe [2^7LiF-BeF_2]) or chloride-based salt, which can be circulating throughout the reactor during operation (GIF, 2021d; Pereira, 2020).

Thermal MSRs are typically designed with graphite as the moderator. Liquid fuels offer the possibility of refueling at power in steady-state operation and continuously removing fission products by salt processing in a chemical processing facility directly attached to the reactor. Liquid fuel thermal MSRs operating in full recycle mode would require either continuous or batch salt processing to remove parasitic fission products with large neutron capture probabilities to maintain desired reactivity. Salt processing to remove fission products could be carried out with halogenation (fluorination or chlorination) playing a key role in many process flow sheets. For example, uranium in a fluoride salt is UF_4. Upon fluorination, UF_4 reacts with fluorine gas to produce the volatile UF_6 species that enters the gas phase, effectively separating uranium from the salt. The resulting UF_6 stream undergoes purification by selective sorption/desorption reactions to remove fission products that may also form volatile fluorides (e.g., Br, I, Cs, Se, Mo, Tc, Ru, Sb, Zr, and Te). The purified UF_6 is reduced to UF_4 prior to being reintroduced as fuel into the reactor. Materials compatibility issues continue to be important challenges for both the halogenation and reduction steps in salt processing. Fuel salt cleanup is also an important process supporting MSR operations, as it addresses such impurities as oxygen to ensure the appropriate redox potential is maintained in the salt. The redox potential is an important factor for controlling corrosion and the salt chemistry.

One advantage of MSRs is that the fuel could be processed during reactor operations in an appropriately sized facility directly attached to the reactor, which would eliminate the need to transport spent fuel to an off-site processing facility. Such a facility would also be substantially smaller with flow rates orders of magnitude lower than that of a centralized processing facility like those used to support solid fuel technologies. Additional advantages to MSRs include that there are no neutron losses in structural materials; there is online refueling; there is no need for solid fuel fabrication; the reactor can operate under low pressures and could go to high temperatures; the reactor has a very low excess reactivity; and the molten salts will not react dangerously with air or water—although, as noted below, there are concerns about corrosion. Disadvantages include material degradation; need for safe management of fission product off-gases, as well as tritium, which would be produced if lithium is used in the salt; need for remote maintenance; difficulty with material inventory tracking; need for electric heaters so

[13] Liquid-fueled MSRs will operate on a modified open fuel cycle that requires an off-gas treatment system to capture and contain volatile fission products. An open fuel cycle in the sense of once-through use of fuel (i.e., no recycle) is possible if the spent fuel salt removed from the reactor is discarded and not treated to recover actinides for reuse as fuel. Volatile fission products may be evolved and captured.

that the salt remains a liquid; and need for both the nuclear plant and online fuel processing plant to maintain high availability simultaneously.

In fast MSRs, losses due to parasitic fission products would be considerably fewer, and salt processing could be carried out less frequently. In a recent report on molten chloride fast reactors, the Electric Power Research Institute found that the fuel salt sustainability over time may be limited by the buildup of fission products, which could "potentially impact reactor performance, waste management, and downstream reuse of fuel salt to start up subsequent units" and cautioned that "an integrated understanding of fuel salt sustainability early in the liquid-fuel MSR design process may help developers and owner-operators avoid downstream design and operational challenges" (EPRI, 2021a). Thus, the online fuel processing needs for fast MSRs would primarily be fission off-gas treatment and the removal of particulates of noble metal fission products that build up in the salt. Fuel cycles that support fast MSRs depend on whether, for example, the reactors are operated as transuranic burners using spent LWR fuel as feed, uranium/plutonium breeders, or converters using natural uranium fuel feed with removal of noble metal fission products by filtration (minimal separations).

Fuel salt chemistry in MSRs is complex, and the salt composition (e.g., quantities of fissile and fertile isotopes, as well as fission, transmutation, and radiolysis products) is continually changing with time. Salt cleanup, salt processing (using, e.g., volatility, reductive extraction, or electrochemical methods), and off-gas treatments are important areas of R&D for MSRs going forward if they are to realize their potential of fully closing the fuel cycle (Fredrickson et al., 2018; Pereira, 2020). Additional R&D challenges relate to the chemical reactivity of molten salts—specifically, chemical compatibility between the molten salt and reactor structural materials, including alloying elements; the potential for enhanced corrosion in the presence of air, moisture, or other nonmetal impurities; and the effects of irradiation (McFarlane, 2021). Further R&D is also needed for waste management—in particular the processing of chloride and fluoride salts required to generate suitable waste forms (McFarlane, 2021). Other key research areas include industrial-scale, economical, and environmentally friendly methods for the isotopic separation and recovery of chlorine-37 and lithium-7 for molten chloride and fluoride FLiBe salt systems, respectively (Arm et al., 2020). Molten salts must be highly enriched in chlorine-37 and lithium-7 to maintain the neutron economy within the reactor and avoid neutron capture reactions that produce large quantities of chlorine-36 (and sulfur-35) and tritium as waste products.[14] Examples of liquid fuel molten salt reactor designs under development include Terrestrial Energy's IMSR-400, ThorCon's ThorCon Reactor, Flibe Energy's Liquid Fluoride Thorium Reactor, Moltex's Stable Salt Reactor-Wasteburner, and TerraPower's Molten Chloride Fast Reactor.

Fluoride-cooled high-temperature reactors (FHRs) are thermal spectrum reactors that are sometimes referred to as MSRs because they use a molten fluoride salt as the coolant; however, FHRs are more properly classified as high-temperature gas-cooled reactors because they use solid fuel, typically TRISO, are graphite moderated, and are designed to operate on an open fuel cycle with no reprocessing (MIT, 2018). Because of the much higher heat capacity of a molten salt compared with a gas such as helium, the use of a molten salt as the coolant provides a significant advantage for removing heat from the reactor core. Research needs for FHR fuel cycles are analogous to those described above for both TRISO fuel and molten salts, and are detailed in Chapters 4 and 5.

Kairos Power is developing an example of this reactor type, the KP-FHR. The company initiated preapplication review activities with the U.S. NRC in September 2018 and subsequently submitted a total of 11 topical reports, 8 of which had received draft or final safety evaluation reports as of May 2022. In November 2021, the U.S. NRC formally accepted its construction permit application for the Hermes low-power test reactor, which will be built at the East Tennessee Technology Park Heritage Center site in Oak Ridge (Kairos, 2021). Kairos aims to have Hermes operational by 2026. Kairos is implementing an iterative development process from engineering test unit to demonstration reactor to commercial scale. In January 2021, Kairos representatives told the committee that they are focused on iterative testing of hardware because it can demonstrate real-world performance unlike relying on computer calculations alone (Blandford and Peterson, 2021). The electrically heated Engineering Test Unit is the first major integrated hardware iteration that has allowed Kairos to demonstrate systems in a nonnuclear environment. Kairos' "iterative strategy is supported by capabilities in material testing, tritium testing, chemistry control, mod/sim [modeling/simulation], core design and neutronics, and instrumentation and controls" (Blandford and

[14] Chlorine-36 is a long-lived ($t_{1/2}$ = 301,000 years) energetic (709 keV) beta emitter that is highly soluble (mobile) in water.

Peterson, 2021). Experience with Hermes would lead to a larger-scale nonnuclear electrically heated integrated demonstration facility, whose testing results would feed into the deployment of the first commercial-scale reactor unit. This KP-X reactor is planned to be operational before 2030, enabling subsequent commercial deployment of additional KP-FHRs in the 2030s (Blandford and Peterson, 2021).

Terrestrial Energy is pursuing licensing applications in the United States for its MSR design, having been in prelicensing application discussions since late 2019 with the U.S. NRC (2022b). Additionally, Terrestrial Energy, ThorCon, and Moltex are pursuing licensing applications outside of the United States. In 2017, Terrestrial Energy completed the first phase of prelicensing review of its IMSR by the Canadian Nuclear Safety Commission (CNSC), which determined that the safety features met the threshold to move toward formal licensing review, and they expect to complete the second-phase licensing review in 2022 (NEI, 2021a). In May 2021, Moltex completed Phase 1 of CNSC prelicensing vendor design review (Moltex Energy, 2021). ThorCon is planning to first build a test reactor outside the United States to gather test data and has signed a memorandum of understanding with Indonesia (Jorgensen, 2021). A ThorCon representative told the committee in January 2021 that the licensing process in the United States is too lengthy and prohibitive for private investors. The same representative noted that, in particular, in the U.S. regulatory process, validated software is needed to license a design before one can build a test reactor, but to validate the software, one needs to use test data.

3.2.3.6 Small Modular Light Water Reactor: NuScale Power Module (iPWR)

Small modular LWRs, such as the NuScale Power Module, are derived from the current commercial large (>300 to 1,000+ MWe) pressurized LWRs. As such, they operate with a thermal spectrum and uranium dioxide fuel enriched to less than 5 percent uranium-235, using light water as both the coolant and moderator. However, in addition to the difference in power rating, these small modular LWRs have other key differences from large LWRs. A small modular reactor "can be constructed and operated in combination with similar reactors at a single site" (Infrastructure Investment and Jobs Act of 2021, Public Law 117-58). Integral designs, called "iPWRs" (integral pressurized water reactors), enclose the whole reactor primary circuit—including the pressurizer, coolant pumps, and steam generators—in a single containment vessel, which allows for the reactor to be compact.

The NuScale Power Module (NPM), an example of an iPWR, is designed to provide 77 MWe per module. The NPM's simple design eliminates the coolant pumps, large-bore piping, and other components typically found in large, conventional reactors. As such, the NPM is small enough to be factory built, enabling easy transport and installation, which could reduce the time and cost of construction (Arostegui and Holt, 2019).[15] The NPM is also scalable for flexible integration to match load requirements (4, 6, and 12 modules per plant). Given the similarity of its base technology to commercial LWRs, the fuel cycle facilities required to fuel and manage the wastes for these reactors—with the exception of a deep geologic repository—already exist. In addition, the NPM is the first small modular reactor to undergo the U.S. NRC licensing process. In September 2020, the U.S. NRC issued a standard design approval to NuScale approving the safety aspects of a 50-MWe version of the NPM design. NuScale will also have to obtain a separate standard design approval for its 77-MWe version, which it plans to deploy in a six-module reference plant to be operational by 2029 at Idaho National Laboratory (Patel, 2022). Of the reactors evaluated by this committee, NuScale's NPM is the only technology that could likely be commercialized in this decade. Examples of other small modular LWRs under development include GE-Hitachi's BWRX-300 (a boiling water reactor) and Holtec International's SMR-160 (a pressurized water reactor). The ground laid by NuScale would likely aid the development and potential deployment of these reactors.

As to technology development, the NPM is nearing commercial deployment by the end of the decade because of several factors: (1) technology derived from existing LWRs; (2) use of uranium dioxide fuel with enrichment less than 5 percent uranium-235, which can be produced with the existing LWR fuel cycle infrastructure; (3) substantial financial support from government and investors; and (4) existing U.S. nuclear regulatory processes for LWR technology. NuScale has received funding for over a decade from several federal awards and especially

[15] These aspects will be covered in more depth in the parallel National Academies study Laying the Foundation for New and Advanced Nuclear Reactors.

from DOE through the SMR Licensing Technical Support Program (which ran from fiscal year [FY] 2012 to FY2017), from DOE-NE's U.S. Industry Opportunities for Advanced Nuclear Technology Development in 2018, and through the Advanced SMR R&D Program in FY2021. NuScale also benefitted from DOE's 2020 award to Utah Associated Municipal Power Systems for the Carbon Free Power Project, which intends to use NuScale's NPM for generation of electrical power. As of end of 2021, NuScale had received more than $1 billion in federal funding through 30 awards (Subsidy Tracker, 2021).

3.2.4 Fuel Cycle Options for Advanced Reactors

During the information-gathering meetings, almost all developers told the committee that they are planning on an open, once-through fuel cycle at least for the near to intermediate terms, and likely for the next 30 years (see Table 3.1). Yet, several developers noted that there is potential in the longer term for their technologies to recycle spent nuclear fuel.

Some developers mentioned potential reprocessing and recycling in terms of proliferation resistance. At the January 2021 meeting, the ThorCon representative said that the company would prefer to recycle the enriched uranium because HALEU is a valuable material. However, the NNSA has advised them "not to push" for recycling of uranium because it is a "touchy subject." Thus, the representative said that ThorCon plans to store the material until the company can work through the bureaucratic issues (Jorgensen, 2021). TerraPower's website states that "Natrium plants will not require reprocessing and will run on a once-through fuel cycle that limits the risk of weapons proliferation" (TerraPower, 2021a). In contrast, however, the General Atomics representative discussed how the company's EM^2 technology can recycle in a proliferation-resistant manner, so as not to require enrichment after the first core loading, saying that "subsequent dry recycle processing eliminates that need for further enrichment" (Back and Schleicher, 2021). See Section 3.2.3.2 for more information about the EM^2 technology's development.

The reactor developers considering recycling also described the purported benefits of reduced costs, reduced waste volumes, decreased radiotoxicity,[16] and improved resource sustainability. TerraPower plans to use a once-through cycle and estimates that up to 30 times better uranium utilization is possible than for LWRs, allowing long-term stability without the need for reprocessing (Hejzlar, 2021). However, the TerraPower representative told the committee at the February 2021 meeting, "If recycling is desired, Natrium can support it at reduced cost due to less frequent reprocessing" (Hejzlar, 2021). While ARC plans to use the once-through cycle, the company's representative noted at the February 2021 meeting that their reactor's "system can be sustained indefinitely if recycling is used" (Sackett and Arthur, 2021). Reprocessing is an option with ThorCon's technology concept to improve resource utilization, but the company does not plan to pursue it in the near to intermediate terms. Oklo is the one developer that has moved forward with plans for recycling and stated at the committee's February 2021 meeting that fast reactors using metallic fuel clearly have potential for recycling (DeWitte, 2021). However, given Oklo's plans for 20-year core development, the reprocessing, if it occurs, would not be in the near term. Additionally, in January 2022, the U.S. NRC denied Oklo's license application, although the company stated that it plans to respond to U.S. NRC's concerns and resubmit its application.

Given the stated positions of almost all developers about pursuit of the once-through cycle for the foreseeable future, the committee chose to emphasize this option for the classes of advanced reactors considered in this study. However, because, as noted above, a number of developers discussed the potential of recycling and ultimately closing the fuel cycle through multirecycling, the committee also considered the possibility of reprocessing and multirecycling. Building on the three main fuel cycle options outlined in NEA-OECD (2021), the committee analyzed for uranium-based fuels (1) the once-through cycle for LWRs; (2) monorecycling of uranium and plutonium as mixed oxide fuel in LWRs; and (3) multirecycling of uranium, plutonium, and minor actinides (americium, curium, and neptunium) in fast reactors, in order to evaluate the merits and viability of different fuel cycle options (see Chapters 2 and 4). Chapter 5 discusses waste management and disposal issues, including those

[16] As measured by the decreased inventory of long-lived transuranic element radioisotopes remaining in the high-level waste.

related to reprocessing, for four representative advanced reactor designs: integral pressurized water reactors, high-temperature gas-cooled reactors, sodium-cooled fast reactors, and molten salt reactors.

The committee also heard from two reactor developers, Flibe and ThorCon, who intend to use a fuel cycle based on thorium rather than uranium. Terrestrial Energy also noted the possibility of operating their reactor as a thorium breeder in the future. In Section 3.2.5, a brief comparison of thorium-based fuel cycles (thorium/uranium) with uranium-based fuel cycles (uranium/plutonium) is provided, noting the potential advantages and challenges of switching to thorium-based fuels. More detailed information regarding the nonproliferation implications of thorium fuel cycles is discussed in Chapter 6.

In summary, whatever fuel cycle option is potentially pursued will require relatively long time periods to implement. According to Dr. Terry Todd, who has almost 40 years of experience in fuel cycle R&D and was the national technical director of DOE-NE's Material Recovery and Waste Form Development program from 2008 to 2020, it will require "15-20 years to design, build, permit, and start up a new fuel cycle facility" (Todd, 2021). Demonstrating the start of a new fuel cycle could just require deploying the first commercial reactor using that option, which could be done in several years potentially. Fully implementing a fuel cycle, however, could require a couple of decades to a century in order to transition from a fleet using one fuel cycle, such as the once-through LWR cycle, to a fleet of fast reactors using multirecycling (Williamson and Taiwo, 2021). To keep options available for potential future fuel cycles will require sustained investments in R&D and workforce development (Zhan et al., 2021). As mentioned in Section 3.2.2, collaborations with international partners and these partnerships could help with cost-sharing in activities such as supporting joint R&D projects and building test reactors for qualifying fuels and testing materials (NEA-OECD, 2018a, n.d.). However, funding at the U.S. national level is still limited, and not all advanced reactor programs can have full government support; thus, decisions will have to be made as to what R&D support to prioritize (Todd, 2021).

3.2.5 Comparison of Thorium-Based Fuel Cycles (Th/U) with Uranium-Based Fuel Cycles (U/Pu)

In the periodic table, only two elements above bismuth (element 83) exist in nature to a significant extent: thorium (element 90) and uranium (element 92). As shown in Table 3.2, thorium has only one long-lived isotope,

TABLE 3.2 Th and U Long-Lived Isotopes (Isotope with Half-Life Greater than 1×10^5 Years)

Atomic Number	Isotope	Natural Abundance (atom %)	Half-Life (years)	Fissile/Fertile
90	^{232}Th	100	1.40×10^{10}	fertile
92	^{234}U	0.01	2.46×10^5	fertile
92	^{235}U	0.72	7.04×10^8	fissile
92	^{238}U	99.27	4.47×10^9	fertile

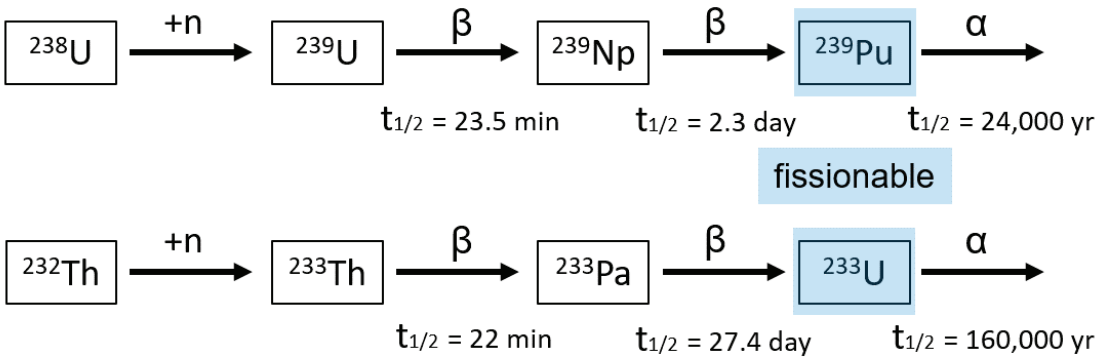


FIGURE 3.2 Pathways for producing ^{239}Pu from ^{238}U (top) and ^{233}U from ^{232}Th (bottom) by the capture of neutrons.

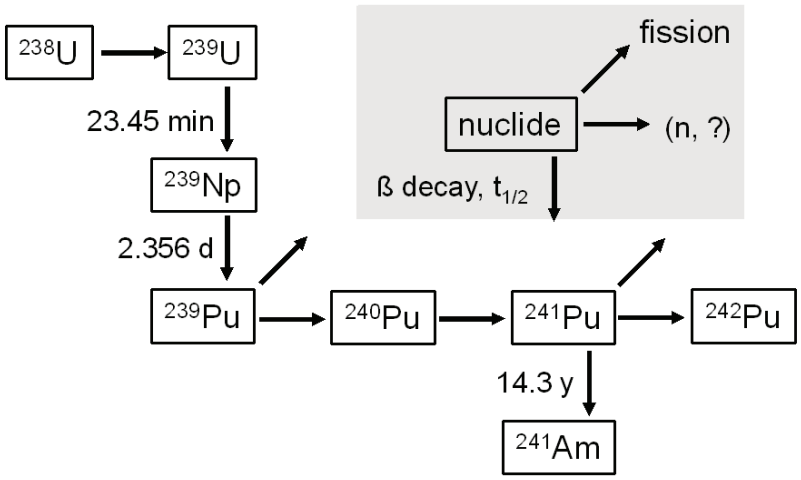


FIGURE 3.3 Successive capture (n,γ)/fission/beta decay reactions originating from a single radionuclide, illustrating generation pathways for the production and destruction of Pu isotopes from the reaction of ^{238}U with neutrons.
SOURCE: Adapted from Murray and Holbert (2014).

while uranium has three. Of these four nuclides, only ^{235}U is fissile; that is, ^{235}U can readily undergo neutron-induced fission and thus be used to fuel a fission reactor. The other three isotopes are termed fertile nuclides and can be converted into fissile nuclides upon neutron capture. Figure 3.2 shows the pathways for producing ^{239}Pu from ^{238}U (top) and ^{233}U from ^{232}Th (bottom) by the capture of neutrons.

When uranium-based fuel is irradiated in a reactor, the fission of ^{235}U generates on average between two and three neutrons. One of these neutrons reacts with another ^{235}U nucleus to sustain the critical fission reaction, while the excess neutrons are available to react with other nuclei in the fuel. If these excess neutrons react with ^{238}U in the fuel, they breed ^{239}Pu, which can go on to fission or capture a neutron to become ^{240}Pu. Even-numbered atomic mass nuclides of Pu[17] do not fission appreciably with low-energy neutrons, so ^{240}Pu tends to capture a neutron and become ^{241}Pu, which can either fission or capture another neutron becoming ^{242}Pu, and so on. Over time, neptunium, plutonium, and heavier minor actinide isotopes (e.g., Am, Cm) are generated via capture and decay reactions, as illustrated in Figure 3.3.

Pure ^{232}Th-based fuel in a reactor requires the addition of fissile material (i.e., ^{235}U, ^{239}Pu/^{241}Pu, or ^{233}U, called seed fuels[18]) to enable and sustain nuclear fission. Since low-enriched U would be inefficient as a seed fuel, the seed fuel needed is higher-enriched U or Pu. When Pu is used as the seed fuel, the fuel would be similar to conventional mixed oxide fuel, but with ^{238}U replaced by ^{232}Th. The ^{239}Pu/^{232}Th fuel is referred to as "Th-MOX." Excess neutrons from the fission of the seed fuel can then go on to breed ^{233}U from ^{232}Th (SNETP, 2011).

Table 3.3 shows the values of η, the number of neutrons produced per neutron absorbed in the fuel, for thermal and fast neutrons. These values indicate that breeding using ^{235}U or ^{239}Pu in a thermal reactor would be difficult, since expected losses to nonbreeding processes, such as absorption, by structural materials would reduce the excess neutrons to one or less. However, with a $\eta = 2.3$ the prospects for breeding ^{233}U from ^{232}Th with thermal neutrons are more reasonable. Comparing all values for η suggests that the most promising breeding occurs with a fast reactor and ^{239}Pu (i.e., a U-based fuel cycle). Once any breeder reactor goes critical, it can, in principle, breed as much or more fissile material than it consumes. Because of the more favorable breeding economy in a fast reactor, once criticality is achieved, a fast breeder can make greater quantities of fissile material than a thermal breeder reactor.

[17] The sentence was modified following release of a prepublication version of the report to clarify that it is referring to Pu specifically, not all even-numbered atomic mass nuclides.

[18] A closed Th fuel cycle in equilibrium would utilize only ^{233}U. However, ^{233}U must first be produced from ^{232}Th, which requires another seed fuel.

TABLE 3.3 Number of Neutrons Produced per Neutron Absorbed in the Fuel, η, for Selected Fissile Isotopes as a Function of Neutron Energy, Thermal (<100 eV) and Fast (>0.1 MeV)

	Neutron Energy	
Isotope	Thermal	Fast
^{235}U	2.07	2.3
^{239}Pu	2.11	2.7
^{233}U	2.30	2.45

SOURCE: Adapted from Murray and Holbert (2014).

3.2.5.1 Advantages and Challenges of a Th/U Fuel Cycle

Th is more abundant in the earth's crust than U by a factor of about 3.3; however, the known reserves of economically extractable Th and U are similar (Touran, 2020). Seawater contains significantly more dissolved U than Th, but predicted costs for U recovery from seawater are not competitive with extraction of natural U deposits by mining (Dungan et al., 2017). With the vast amount of ocean water (about 300 million cubic miles) there are about 4 billion tons of uranium in the ocean at any given time (NEA and IAEA, 2020). Although advances continue to be realized for the extraction of U from seawater, at this time the technology is not commercially viable on an industrial scale (Berger, 2018). In the future, this technology could become viable with advances in U absorbent and fiber technology (Szondy, 2018; Xu et al., 2020a).

Because η is greater than 2 over a wide range of neutron energies, the $^{232}Th/^{233}U$ fuel cycle can operate as a breeder reactor with fast, epithermal, and thermal neutrons, whereas breeding is possible only with fast neutrons for the $^{238}U/^{239}Pu$ fuel cycle. The thermal neutron absorption cross section is three times higher with ^{232}Th than with ^{238}U.

Potential benefits and challenges of implementing a Th fuel cycle are discussed in IAEA (2005a) and summarized here. If the fuel is reprocessed from a thermal reactor operating on a Th/U fuel cycle, mining of U for ^{235}U can be eliminated, thus extending nuclear fuel resources by two orders of magnitude without the need to deploy fast reactors. Breeding ^{233}U from ^{232}Th over time also can eliminate the need to enrich uranium as part of the fuel cycle.

The Th/U fuel cycle does not produce appreciable transuranium (TRU) elements, since ^{238}U is not a starting material. The Th/U fuel cycle produces some ^{237}Np, but if the Np is removed via chemical separation, no Pu can be produced during irradiation. By avoiding Pu altogether, some consider the Th/U fuel cycle more proliferation resistant than U/Pu cycles. However, since ^{233}U and ^{239}Pu have similar nuclear properties, these fuel cycles have comparable proliferation risks. One option for reducing proliferation concerns is to add sufficient U to the initial Th load so that the ^{233}U is isotopically diluted (denatured) with ^{238}U. Isotopic separation would be required to obtain special nuclear material at the expense of the production of more TRU; such an option negates the original advantage of avoiding Pu using Th-based fuels.

There has been considerably less experience with Th than with U/Pu fuel cycles. In the mid-1950s, Oak Ridge National Laboratory developed the THORium-uranium EXtraction (THOREX) process for reprocessing of spent Th–based fuels (IAEA, 2005a). Similar to PUREX, THOREX is a solvent extraction–based separation of Th from fission products by means of tributyl phosphate (TBP). The chemistry of Th for nuclear power–generation applications (the THOREX process) is much less mature than the PUREX process for U and has not been performed commercially. Reprocessing using THOREX has been carried out in only a few countries and mostly in laboratory or pilot plant scale (Balakrishna, 2012; IAEA, 2005a). ThO_2 is chemically more inert (more stable) than UO_2 and as such, ThO_2-based fuel does not easily dissolve in concentrated nitric acid without the addition of HF, which leads to corrosion of process equipment.

Because the melting temperature for ThO_2 (3,390°C) is much higher than that of UO_2 (2,865°C), producing high-quality (high-density) solid Th fuel requires higher sintering temperatures compared with solid U fuel. However, ThO_2-based fuels have a higher radiation resistance, 10 times lower fission product release rate, higher thermal conductivity, and lower coefficient of expansion compared with UO_2-based fuels. For these reasons, ThO_2-based fuels have better in-pile performance than UO_2 or UO_2–mixed oxide fuels (suggesting that higher burnups can be achieved with Th-based fuels).

During the THOREX process, protactinium is normally discarded as waste with the fission products. Proliferation concerns of the Th/U fuel cycle center around the separation of ^{233}Pa as an intermediate isotope.[19] Since ^{233}Pa has a relatively long half-life (27 days) compared with ^{239}Np (2.35 days) in the uranium fuel cycle, a cooling time of about a year (more than 10 half-lives of ^{233}Pa to allow the decay of ^{233}Pa to ^{233}U) is necessary prior to reprocessing to avoid loss of ^{233}U fissile material (IAEA, 2005a). The diversion of separated ^{233}U during fuel reprocessing poses a proliferation risk that can be mitigated by adding ^{238}U (denaturing) (IAEA, 2005a).

In Th-based fuels, ^{232}U (half-life 73.6 years) is produced by (n,2n) reactions with ^{232}Th, ^{233}Pa, and ^{233}U. The short-lived daughter products of ^{232}U, such as ^{212}Bi and ^{208}Tl, emit strong gamma radiation in Th/U fuel that requires more shielding, making handling and reprocessing more difficult but not impossible, and more expensive compared with U/Pu fuel.

^{232}Th blankets can be used to breed ^{233}U, in a manner analogous to breeding Pu using ^{238}U in "conventional" fast breeder reactors.

3.2.5.2 Reactors Capable of Using Th Fuel

Six different reactor types can use Th as a fuel: pressurized heavy water reactors, high-temperature gas-cooled reactors (HTGRs), boiling water reactors, pressurized light water reactors, fast neutron reactors, and molten salt reactors (WNA, 2020c). Some demonstration reactors that have used Th-based fuels to generate electricity include the pebble-bed 300-MWe Thorium High-Temperature Reactor in Germany, and the 40-MWe Peach Bottom HTGR and 330-MWe Fort St. Vrain HTGR in the United States. For all three of these reactors, the reference fuel was Th-HEU (highly enriched U). Because of proliferation concerns, the use of HEU (U enrichment to greater than 20 percent) in civilian reactors over the past several decades has been significantly curtailed worldwide, and implementing a Th fuel cycle with HEU seed fuel would not be considered today. Furthermore, replacing the reference HEU fuel with <20 percent low-enriched U considerably reduces its overall performance, making this Th fuel cycle much less attractive (NEA-OECD, 2015a).

The first reactor to use ^{233}U as fuel was the Molten Salt Reactor Experiment at Oak Ridge National Laboratory in 1968. In the late 1970s, breeding ^{233}U was demonstrated in a thermal LWR at the Shippingport Atomic Power Station in Pennsylvania (Olson et al., 2002).[20] The Shippingport reactor core consisted of 12 seed modules, containing >98 percent ^{233}U, surrounded by a stationary blanket module. To achieve breeding, the fuel assemblies were moved through the core to adjust reactivity rather than using control rods. Also, the fuel had to be removed and processed at low burnup, where the conversion ratio is optimal. Consequently, breeding ^{233}U in a thermal spectrum would require specially designed reactor systems that are different from the current commercial LWRs. Other reactor types are better suited to the thermal breeding of ^{233}U (Wigeland et al., 2009). In the mid-1990s, there was renewed interest in Th fuel cycles in the context of advanced reactor development, and in particular for molten salt reactors.[21] Solid fuel is not an issue in molten salt reactors, making them best suited for thermal ^{233}U breeding (GIF Experts' Group, 2010). Continuous online processing of molten salt fuel optimizes the overall breeding ratio by removing fission product poisons and separating ^{233}Pa from the core to minimize parasitic neutron capture, which produces ^{234}U, a fertile isotope, at the expense of ^{233}U production (U.S. NRC, 2014d). Molten salt reactors also offer the possibility of adjusting the fertile/fissile fuel composition without shutting down the reactor (NEA-OECD, 2015a). However, the development, licensing, and construction of advanced reactor systems that might realize the full benefit of a closed Th/U fuel cycle is a long-term undertaking, as the requisite dedicated Gen IV and beyond breeder reactors, including molten salt reactors, are currently in the conceptual design phase.

[19] In the Th conversion chain, ^{233}Pa has to be removed from the thermal flux of the reactor; otherwise ^{233}Pa will capture thermal neutrons, becoming ^{234}Pa, and decay within a few hours to ^{234}U, which is fertile, rather than the intended fissile ^{233}U.

[20] Shippingport Atomic Power Station in Pennsylvania (USA) operated from 1977 to 1982.

[21] The use of the MOlten Salt Advanced Reactor Transmuter (MOSART) "concept as a transmuting system cycle initially fed with TRU loading from LWR SNF for producing the necessary fissile ^{233}U quantities for a uranium-thorium fuel cycle" (Ignatiev et al., 2014).

3.2.5.3 Relative Radiological Hazards of Key Isotopes in Th/U and U/Pu Fuel Cycles

The actinides play a dominant role in terms of total radioactivity, ingestion radiotoxicity, and potential dose to the public. A much smaller quantity of Pu and long-lived minor actinides (Np, Am, and Cm) is created in the ^{232}Th/^{233}U fuel cycle as compared with the ^{238}U/^{239}Pu fuel cycle, thereby minimizing toxicity and decay heat problems (IAEA, 2005a). The commonly stated perception that the ^{232}Th/^{233}U fuel cycle will generate less Pu and minor actinides, thereby reducing the radioactive hazard of the spent fuel from the ^{232}Th/^{233}U fuel cycle compared with that of the ^{235}U/Pu fuel cycle, may not be correct. According to Piet (2013), "Thorium/^{233}U fuel cycles do have lower amounts of TRU isotopes, but that does not necessarily mean lower 'long-term hazard' since focusing on TRU without adequate attention to actinide decay products can lead to incorrect conclusions." Furthermore, Piet found that:

> ^{239}Pu and ^{233}U have orders of magnitude per kilogram higher radiotoxicity than the ^{238}U or ^{232}Th from which they were made. Per kilogram of fissile material bred, initially pure ^{239}Pu is slightly more radiotoxic than initially pure ^{233}U until about 40 000 y after reactor discharge. Thereafter, ^{233}U is more radiotoxic than ^{239}Pu, as ^{233}U progeny such as ^{229}Th accumulate. By a few million years, the picture changes again as ^{239}Pu has decayed to ^{235}U and ^{233}U has decayed substantially. Uranium/^{239}Pu and Th/^{233}U both have important actinide isotopes; sometimes they are "trans" the fuel isotopes, such as TRU ~beyond uranium, and sometimes they are just below the fuel isotopes, such as ^{229}Th from Th/^{233}U. Thus, from the standpoint of comparing inventories, one cannot say that that either Th/^{233}U or U/^{239}Pu is necessarily more benign than the other from the standpoint of long-lived environmental burdens. (Piet, 2013)

Croff and Krahn (2016) reached the same conclusion, noting, "When analyzed on a consistent basis (e.g., same reactor design, fuel design, burnup, specific power, cladding type and composition, and fuel matrix trace element concentration), the radiotoxicity of thorium-based fuels and wastes being disposed of in a repository are broadly similar to those for uranium-based fuel and not 'far lower.'" Furthermore, Croff and Krahn (2016) found that "at relatively short times (less than a few centuries) and very long times (greater than a few million years), the radiotoxicities of uranium- and thorium-based SNFs [spent nuclear fuels] are essentially equal."[22] "Overall, the ingestion radiotoxicity of thorium-based fuels containing ^{233}U or plutonium fissile materials is similar to the radiotoxicity of uranium-based fuels containing ^{235}U or plutonium fissile materials for decay times ranging from 1 year to 20 million years" (Croff and Krahn, 2016). Figure 3.4 shows the ingestion radiotoxicity of 1 kg each of ^{232}Th, ^{233}U, ^{235}U, ^{238}U, and ^{239}Pu and their progeny as a function of time. A report by the Nuclear Energy Agency of the Organisation for Economic Co-operation and Development on Th fuel cycles highlights several long-lived radionuclides generated in Th fuel cycles—^{233}U, ^{231}Pa, and ^{232}U—that "have a more important radiotoxicity than their counterparts in the uranium cycle," and also concludes that "radiotoxicity of thorium-based fuels is more accurately described as being comparable to that of uranium-based spent nuclear fuel" (NEA-OECD, 2015b).

3.2.5.4 The Bottom Line: Uranium-Based (U/Pu) or Thorium-Based (Th/U) Fuel Cycles?

DOE's Advanced Fuel Cycle Initiative Options Study reviewed and evaluated alternative fuel cycles and technology options, including a comparison of Th/U and U/Pu fuel cycles (Wigeland et al., 2009). The study concluded that "the thorium option would have lower, but probably not significantly lower, TRU inventory and disposal requirements, both having essentially equivalent proliferation risks" (Wigeland et al., 2009). It further concluded that "the choice between uranium-based fuel and thorium-based fuels is seen basically as one of preference, with no fundamental difference in addressing the nuclear power issues," noting, however, that the lack of infrastructure for Th-based fuels in the United States and the lower technical maturity of processing Th versus U fuels would mean higher costs and R&D and demonstration requirements for the Th option. Developing technologies and processes for Th fuels to supplement U fuel technology could enable future nuclear expansion in the United States.

[22] Although minor actinides (Am, Np, Cm) are not produced to any appreciable extent in Th-based fuel cycles, Th-based fuel cycles are associated with relatively short half-life isotopes, such as ^{232}U and ^{228}Th, as well as other radionuclides such as ^{231}Pa, ^{229}Th, and ^{230}U, which have a longer-term radiological impact (IAEA, 2005a).

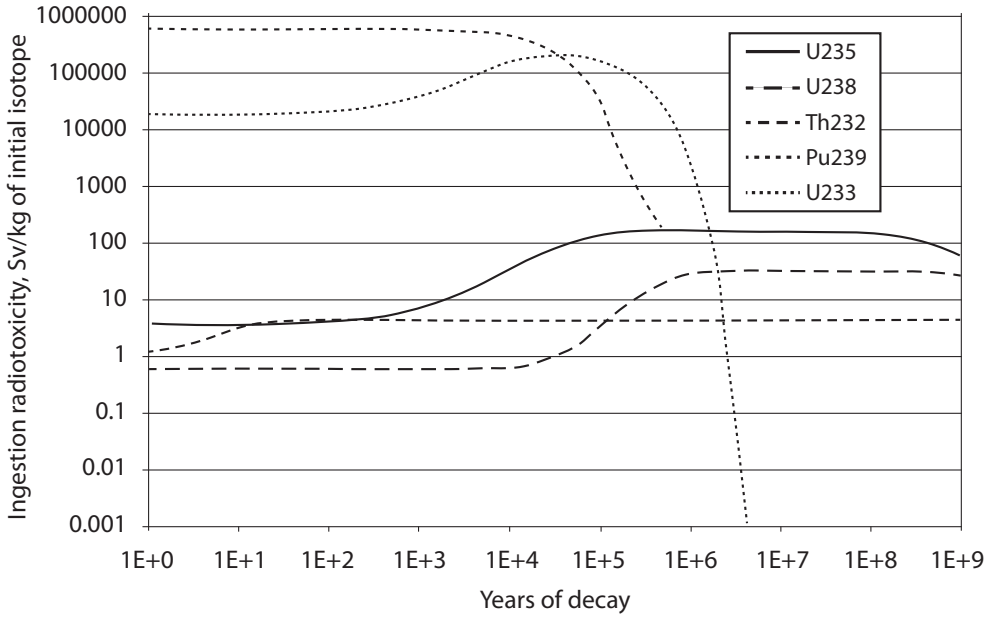


FIGURE 3.4 Ingestion radiotoxicity of 1 kg each of ^{232}Th, ^{233}U, ^{235}U, ^{238}U, and ^{239}Pu and their progeny as a function of time. SOURCE: Piet (2013).

Although most countries have opted for continued development of the U/Pu fuel cycle, India is the exception. India has been actively pursuing a Th fuel cycle because its natural Th resources are larger than its natural U resources. As discussed in Appendix H, India formulated a three-stage program "in the middle of the last century to exploit the full energy potential of its resources" (IAEA, 2008).

> The first stage utilizes the limited natural uranium resources for both power production and the conversion of uranium to plutonium. The plutonium produced in the first stage will form the fuel for the second stage, where it will be used in fast breeder reactors to produce power and enhance the fissile inventory necessary for launching the third stage thorium-based power reactors. Reprocessing and recycling of both fissile and fertile components back into appropriate reactor systems is an integral part of this strategy. (IAEA, 2008)

3.3 U.S. GOVERNMENT SUPPORT FOR DEVELOPMENT OF ADVANCED REACTORS AND ASSOCIATED FUEL CYCLES

Chapter 2 provides information on U.S. government efforts since the early 2000s until about 2017 to reinitiate R&D on advanced reactors and their fuel cycle options. This section focuses on U.S. government efforts since 2017 to develop these reactors and fuel cycles.

3.3.1 Congressional Actions

As introduced in Chapter 1, Congress's enactment of the Nuclear Energy Innovation Capabilities Act in 2018 (NEICA) (Public Law 115-248) defined advanced reactors and their potential improvements in comparison with existing LWRs. NEICA also included a number of provisions regarding testing and demonstration of advanced reactor concepts. In particular, it required DOE to "determine the mission need for a versatile reactor-based fast neutron source, which shall operate as a national user facility." The user facility would "provide at a minimum: (1) fast neutron spectrum irradiation capability, and (2) capacity for upgrades to accommodate new or expanded research needs." In addition, NEICA required DOE to "carry out a program for enhancing the capability to develop

new reactor technologies through high-performance computation modeling and simulation techniques." As to the roles of the private sector, NEICA authorized "a program to enable the testing and demonstration of reactor concepts to be proposed and funded by the private sector" and directed DOE to "leverage the technical expertise of relevant federal agencies and the national laboratories in order to minimize the time required to enable construction and operation of privately funded experimental reactors at national laboratories or other DOE-owned sites."

The Nuclear Energy Innovation and Modernization Act (NEIMA) (Public Law 115-439) was signed into law in January 2019 and reiterated the attributes of advanced reactors, as defined by NEICA. NEIMA's main purpose was "to provide a program to develop the expertise and regulatory processes necessary to allow innovation and the commercialization of advanced nuclear reactors." NEIMA directed the U.S. NRC to develop, within the existing regulatory structure, procedures and processes for licensing of advanced commercial reactors, as well as research and test reactors. Moreover, not later than December 31, 2027, the U.S. NRC is to "complete a rulemaking to establish a technology-inclusive, regulatory framework for optional use by commercial advanced nuclear reactor applicants for new reactor license applications."

The relevant aspects for advanced nuclear reactors of the Infrastructure Investment and Jobs Act (Public Law 117-58), enacted November 15, 2021, are described in Section 3.3.2.

3.3.2 U.S. Department of Energy's Office of Nuclear Energy's Programs and Goals

On January 8, 2021, DOE-NE published its *Strategic Vision* for supporting existing reactors, enabling deployment of advanced reactors, developing advanced fuel cycles, maintaining U.S. leadership in nuclear energy technology, and enabling a high-performing organization (DOE-NE, 2021a). For advanced reactors and fuel cycles, the *Strategic Vision* includes the following goals, objectives, and indicators.

A key goal is to enable advanced reactor deployment to meet these objectives:

- "Reduce risk and time needed to deploy advanced nuclear technology.
- Develop reactors that expand market opportunities for nuclear energy.
- Support a diversity of designs that improve resource utilization."

The deployment performance indicators are

- "By 2024, demonstrate and test a fueled microreactor core fabricated by advanced manufacturing techniques.
- By 2025, enable demonstration of a commercial U.S. microreactor.
- By 2027, demonstrate operation of a nuclear-renewable hybrid energy system.
- By 2028, demonstrate two U.S. advanced reactor designs through cost-shared partnerships with industry.
- By 2029, enable operation of the first commercial U.S. small modular reactor.
- By 2035, demonstrate at least two additional advanced reactor designs through partnerships with industry."

For developing advanced nuclear fuel cycles, the goal is to attain these objectives:

- "Address gaps in the domestic nuclear fuel supply chain.
- Address gaps in the domestic nuclear fuel cycle for advanced reactors.
- Evaluate options to establish an integrated waste management system."

The development performance indicators are

- "By 2021, begin procurement process for establishing a uranium reserve.[23]
- By 2022, demonstrate domestic [high-assay low-enriched uranium] HALEU enrichment.[24]

[23] DOE did not achieve this objective.

[24] As of April 2022, it appeared that this objective's achievement could be delayed. See Chapter 4 for more details.

- By 2023, make available up to five metric tons of HALEU from non-defense DOE material.[25]
- By 2030, evaluate fuel cycles for advanced reactors."

The remaining parts of this subsection describe the DOE-NE programmatic activities to support the above goals, objectives, and performance indicators.

In May 2020, DOE-NE launched the Advanced Reactor Demonstration Program (ARDP) to partner with domestic private industry to demonstrate advanced nuclear reactors. This program includes three levels of funding opportunities:

- Advanced Reactor Demonstration awards, which provided $160 million in initial funding for cost-shared reactor demonstrations to be operational within 5–7 years of the award.
- Risk Reduction for Future Demonstration awards, which provided $30 million in initial funding to support reactor designs that could be operational within 10–14 years of the award.
- Advanced Reactor Concepts 2020 awards, which provided $20 million of funding to help move early-phase reactor designs toward demonstration.

In October 2020, DOE-NE announced the selection of TerraPower and X-energy as the Advanced Reactor Demonstration award winners for demonstration of the Natrium and Xe-100 reactor designs, respectively. On December 16, 2020, DOE announced the selections of five teams to receive funding under the Risk Reduction for Future Demonstration program: Kairos Power, LLC, for the Hermes Reduced-Scale Test Reactor; Westinghouse Electric Company, LLC, for the eVinci microreactor; BWXT Advanced Technologies, LLC, for the BWXT Advanced Nuclear Reactor; Holtec Government Services, LLC, for the Holtec SMR-160 Reactor; and Southern Company Services, Inc., for the Molten Chloride Reactor Experiment. Finally, in late December 2020, DOE announced the three Advanced Reactor Concepts 2020 award winners: Advanced Reactor Concepts, LLC, for the Inherently Safe Advanced SMR for American Nuclear Leadership; General Atomics for the Fast Modular Reactor Conceptual Design; and the Massachusetts Institute of Technology for the Horizontal Compact High Temperature Gas Reactor.

On November 15, 2021, the Infrastructure Investment and Jobs Act authorized $3.2 billion through FY2027 for the Advanced Reactor Demonstration awardees. With the previous authorizations from FY2020 and FY2021, these demonstration projects are fully authorized. In addition, the Act appropriates $2.4 billion from FY2022 through FY2025 for the ARDP's identified existing awardees. This appropriation applies to all parts of the ARDP, including the Risk Reduction and Advanced Reactor Concepts 2020 awardees. The DOE-NE portion to each of the two awardees (TerraPower's Natrium and X-energy's Xe-100) for the Advanced Reactor Demonstration will be $1.89 billion and $1.25 billion, respectively. With the company's private funding included, the total financial support will be approximately $4.0 billion for the TerraPower project and $2.5 billion for the X-energy project over a 7-year period.

DOE-NE's support for advanced reactor development and deployment involves working with and through national laboratory–led R&D, university research programs, and cost-shared private–public industry partnerships. The objectives are as follows:

- "Conduct focused research and development to reduce technical barriers to deployment of advanced nuclear energy systems.
- Develop technologies that can enable new concepts and designs to achieve enhanced affordability, safety, sustainability, and flexibility of use.
- Sustain technical expertise and capabilities within national laboratories and universities to perform needed research.
- Engage with Standards Developing Organizations (SDOs) to address gaps in codes and standards to support advanced reactor designs.

[25] As of April 2022, this objective also appeared unlikely to be achieved.

- Collaborate with industry to identify and conduct essential research to reduce technical risk associated with advanced reactor technologies" (Caponiti, 2020b).

For the specific advanced reactor technologies within the scope of this study, DOE-NE's main research areas are as follows:

- Fast reactor technologies
  - Demonstration of feasibility of advanced systems and component technologies
  - Methods and code validation to support design and licensing
  - Qualification of legacy metallic fast reactor fuel performance data
- Gas reactor technologies
  - Advanced alloy and graphite materials qualification
  - Scaled integral experiments to support design and licensing
  - TRISO-coated particle fuel development and qualification
- Molten salt reactor technologies
  - Investigation of fundamental salt properties
  - Materials, models, fuels, and technologies for salt-cooled and salt-fueled reactors

DOE-NE's Gateway for Accelerated Innovation in Nuclear (GAIN) program provides "access to technical, regulatory, and financial support" for nuclear energy developers, both existing commercial reactor technologies and advanced reactors (Caponiti, 2020b). Situated at Idaho National Laboratory, GAIN and its website provide a number of resources.[26] Notably, GAIN has funding opportunities intended to help industry accelerate deployment of commercial reactors. It can offer access to technical and regulatory support, and has available a legacy U.S. nuclear research database, advanced computational tools, and access to nuclear research expertise. GAIN also convenes several workshops and webinars on topics such as HALEU, fast reactor technologies, and small modular reactors (Caponiti, 2020b).

DOE-NE provides additional U.S. industry opportunities for development of advanced nuclear technologies through its U.S. Industry Opportunities for Advanced Nuclear Technology Development program. These opportunities are intended to support innovative designs and technologies that have significant potential to increase the economic prospects of nuclear power in the United States. DOE-NE expects the resulting products to be manufactured in the United States (Caponiti, 2020b). DOE-NE has two award cycles annually for these opportunities, and the awards are described on DOE-NE's website.[27]

DOE-NE also supports programs to ensure a supply of HALEU; to develop fast reactor fuels; to support R&D on materials recovery and waste forms; to develop materials protection, accounting, and control technology; and to build the Versatile Test Reactor (Griffith, 2020). For domestic HALEU acquisition pathways, the three approaches are (1) downblending to HALEU enrichments from the recovery of high-enriched uranium reprocessed from spent fuel, such as from Experimental Breeder Reactor II, naval reactors, and the Advanced Test Reactor; (2) contracting with Centrus Energy Corporation for the advanced centrifuge demonstration project; and (3) producing HALEU from limited amounts of material in DOE's uranium inventory. More details on HALEU acquisition and supply chain are in Chapter 4.

For developing fast reactor fuels, three notable R&D areas are (1) metallic fuels for closed fuel cycles and actinide transmutation for waste management missions; (2) fuels for once-through fast spectrum reactors; and (3) fuels for high-temperature, fast-spectrum reactors. For R&D on materials recovery and waste forms, four topics of interest to DOE-NE are (1) investigating fundamental fuel cycle chemistry; (2) supporting transformative studies in domestic molten salt processing and chemistry; (3) developing advanced waste forms for aqueous and salt processing; and (4) conducting a joint fuel cycle study with the Republic of Korea "to assess the technical feasi-

[26] See https://gain.inl.gov/SitePages/Home.aspx.

[27] See, for example, awardees for 2021: https://www.energy.gov/ne/articles/doe-awards-85-million-advance-promising-nuclear-technologies.

bility of electrochemical recycling for managing used fuels" (Griffith, 2020). The Advanced Reactor Safeguards Program was also established by DOE-NE in 2020 as part of the appropriations for the ARDP (Cipiti, 2021a).

The Versatile Test Reactor (VTR) project was launched in February 2019 and was "proposed to be a 300 MWth sodium-cooled, fast spectrum reactor capable of testing advanced nuclear fuels and materials for the next generation of nuclear reactors" (Griffith, 2020). In summer 2021, the House and Senate Appropriations Committees zeroed out funding for the VTR facility in FY2022, giving no reason at that time for doing so. DOE has estimated the cost for completing the VTR as between $2.6 billion and $5.8 billion, which would require annual appropriations substantially above DOE-NE's typical budgets. The Biden administration stated that it planned to delay VTR's construction to follow the building of TerraPower's Natrium reactor, which shares similar base technology as the VTR and already has support from DOE's ARDP (AIP, 2021). In an October 2021 interview with *Physics Today*, DOE stated that a plan was being developed and "will address collaborations with industry, such as the VTR/TerraPower [combination], as one of the methods VTR will use to establish public-private partnerships to complete this critical piece of nuclear energy research and development infrastructure" (Kramer, 2021). The final environmental impact statement for the VTR was released in May 2022 (DOE, 2022c). Section 3.4 discusses testing and test reactors in greater detail.

Given budgetary constraints, DOE-NE will need to make difficult decisions in the coming years about its programs. Congressional staff from the U.S. Senate and House Appropriations Committees told the committee in December 2021 that any particular DOE-NE program could consume the entire DOE-NE budget, and there is no feasible way to fully fund, even with industry cost sharing, all the advanced reactors in the ARDP. Congressional staff also said that having a diversity of reactors in the ARDP portfolio is a useful competitive market approach, but industry cannot rely solely on government support. Government support has the important role of maintaining continuity of R&D across programs (Goldner and McKee, 2021). However, constrained budget environments are likely to persist for the coming years. At the same meeting, Dr. Kathryn Huff, then–principal deputy assistant secretary for DOE-NE, acknowledged that DOE-NE has competing priorities and provided her view that, while continued R&D in reprocessing is important for keeping U.S. fuel cycle options open, cost reductions are needed because reprocessing is presently too expensive (Huff, 2021).

3.3.3 The Nuclear Energy Advisory Committee

In 1998, the Nuclear Energy Advisory Committee (NEAC) was established to provide independent advice to DOE-NE "on complex science and technical issues that arise in the planning, managing, and implementation of DOE's nuclear energy program" (DOE-NE, n.d.-b). Operating in accordance with the Federal Advisory Committee Act, NEAC has a diverse membership with U.S. and foreign experts from industry, national laboratories, and universities. NEAC has periodically reviewed DOE-NE's programs and has provided advice and recommendations on DOE-NE's plans, priorities, and strategies for addressing the scientific and engineering challenges of R&D efforts (DOE-NE, n.d.-b). In addition, the secretary of energy or the assistant secretary for DOE-NE can request NEAC to provide advice on national policy and technical aspects of pending DOE decisions on nuclear energy programs. NEAC forms subcommittees to address specific issues related to nuclear energy. The Advanced Reactor Pipeline Subcommittee's objective is to provide an independent and expert review of efforts within DOE-NE related to advanced reactors (DOE-NE, 2019). In December 2021, NEAC's charter was renewed by DOE for another 2 years (DOE, 2021). In February 2022, 11 new members were appointed to NEAC, and DOE-NE announced that it will revise the existing structure in order to have NEAC focus on advising "on current priorities rather than reviewing projects and initiatives that have already been completed" (DOE-NE, 2022a).

3.3.4 U.S. Nuclear Regulatory Commission's Regulatory Programs on Advanced Reactors and Associated Fuel Cycles

As mentioned above, NEIMA directs the U.S. NRC to develop clear regulatory procedures for licensing commercial advanced reactors, as well as for test and research reactors and facilities. As discussed in Section 2.5.1 of Chapter 2, from 2008 to 2016 the U.S. NRC examined potential rulemaking for a commercial reprocessing

facility but suspended this activity in 2016 because of limited industry interest. Section 3.2.3 provides additional information about the interest expressed by advanced reactor developers in reprocessing and recycling. Furthermore, as discussed in Chapter 6, the U.S. NRC is considering options for resuming a rulemaking for "enhanced security" of nuclear materials that was suspended in 2016. If developed, this rule would update requirements and address gaps for a range of nuclear materials that may be used in advanced reactor fuel cycles.

During its September 2020 information-gathering meeting, the committee received briefings from the U.S. NRC staff on the regulatory programs relevant for advanced reactors and associated fuel cycles (Regan et al., 2020). The U.S. NRC's preparations for regulating advanced reactors and fuel cycles include both front-end (especially fuel supply) and back-end (especially spent fuel management) considerations. Also, the U.S. NRC staff underscored that "existing risk-informed and performance-based regulatory framework for licensing and oversight has sufficient flexibility to ensure the safe and secure operation of the complete fuel cycle for advanced reactors." Moreover, the U.S. NRC staff are continually coordinating within their agency and with other government agencies, such as DOE, to collect relevant information and gain insights as to industry developments. Potential licensee applicants are encouraged "to engage early and often in the interest of ensuring complete, high-quality license applications" (Regan et al., 2020).

The U.S. NRC has significant experience in licensing several nuclear fuel cycle facilities, notably for the existing LWR fleet. This regulatory framework also lays the foundation for the regulatory process for advanced reactors' fuel cycles, but the advanced reactor systems could pose challenges different from those of the LWR systems. Both ARDP awardees have engaged the U.S. NRC in preapplication discussions associated with their nuclear plant designs. In addition, U.S. NRC staff have conducted preapplication meetings with one developer (X-energy) to discuss plans for fuel fabrication in a facility licensed by the U.S. NRC. In April 2022, X-energy submitted an application for a Category II license to fabricate TRISO-coated particle fuel forms using up to 19.75 percent enriched uranium. In addition, the U.S. NRC has contracted with the national laboratories to identify potential hazards associated with metallic fuel fabrication and fuel salt preparation for molten salt reactors (Regan et al., 2020). The U.S. NRC's security and safeguards activities related to advanced reactors' fuel cycles, especially use of HALEU, are covered in Chapter 6.

Regarding disposal of high-level waste and spent fuel from advanced reactors, the U.S. NRC staff were not "aware of any technical issues that would require changes to its [the U.S. NRC's] disposal safety requirements to accommodate other fuel types and waste forms" (Regan et al., 2020), and the U.S. NRC is ready to support the national program when given statutory direction. In addition, the U.S. NRC has experience in approval of transportation packages and storage systems for TRISO and metallic fuels. Moreover, the U.S. NRC is "completing technical evaluations on transport, storage, and disposal activities of advanced reactor fuel designs to identify potential information needs and determine whether additional updates to safety review guidance may be warranted" (Regan et al., 2020). Chapter 5 provides more detailed information about transportation, storage, and disposal aspects of advanced reactors and their fuel cycles.

3.4 PROTOTYPING, TESTING, AND TEST REACTORS

The following sections describe the role of prototype and test reactors, as well as other types of test facilities, in supporting R&D on advanced reactors and fuels, and provide the committee's views on this needed infrastructure. According the U.S. NRC, a prototype plant is a nuclear reactor used for testing design or safety features. Such a plant could have "additional safety features to protect the public and the plant staff from the possible consequences of accidents during the testing period" (10 CFR § 50.2; U.S. NRC, n.d.).[28] A test reactor could be a smaller-scale version of an advanced reactor for the purposes of providing data for U.S. NRC testing requirements, proving the concepts of the advanced reactor's materials, components, structures, and various systems. Testing facilities for reactors or other neutron sources can be used to evaluate the behavior under irradiation of fuels and other materials intended for use in advanced reactors (U.S. NRC, n.d.).

[28] This is discussed in the 2007 rulemaking amending 10 CFR § 52 (U.S. NRC, 2007) and in U.S. NRC (n.d.).

3.4.1 Prototyping

Prototyping is a necessary and expensive activity of new reactor and fuel cycle development. Effective prototyping will speed development efforts; improve modeling codes for specific reactors industry wide; reduce costs through the elimination of unknowns and associated conservatism; and provide for increasing regulatory confidence, which is necessary to translate into the savings that so many developers are anticipating. Prototyping is an important part of performance validation for materials, components, reactor systems, and accident scenarios.

Much of the original material, fuel, and component testing for the LWR commercial nuclear industry was performed by government subsidized and operated reactors, including the Shippingport demonstration project in the 1950s. DOE is currently pursuing a similar approach for advanced reactors by proposing construction of the VTR and cosharing the development (including prototyping) expenses with some advanced reactor developers. See Section 3.3.2 for discussion of the VTR program.

Effective prototyping for any nuclear reactor system would meet several needs:

- integrated system performance;
- ability to achieve servicing and inspection improvements;
- proof—under a variety of accident scenarios—of innovative safety improvements, such as natural circulation heat removal; and
- validation of design models.

Notably, advanced reactors, which propose using a wider variety of coolants and fuel types, will have more comprehensive prototyping requirements. In addition to the prototyping described above, advanced reactor developers will have to do substantial prototyping of basic materials' strength and performance when subjected to entirely different operating environments, including differing coolants, neutron flux, temperature, and pressures. Much of this basic prototyping can be accomplished in test reactors where available.

3.4.2 Testing and Test Reactors

Test reactors are a part of the critical infrastructure of the nuclear industry worldwide. They are essential for testing fuel and fuel burnup, materials and welds under a variety of operating environments, operating conditions, and the development of increasingly complicated computer codes used industry wide. They enable innovation and refinement of engineered systems and components and are key to projecting performance for advanced reactors that inherently lack operational data to validate system and component performance under the specific conditions of the advanced design.

Test and operational performance data are critical for understanding material and fuel performance, developing operating parameters and regulatory infrastructure, manufacturing higher-performance and longer-lasting components, and determining inspection regimes. These data are used by research institutions, government laboratories, regulatory agencies, and manufacturers to drive the innovation needed to make the advances necessary to overcome the wide array of challenges faced by industry in successfully deploying advanced designs. Welds, for example, are tested under compression and tension, at different temperatures and pressures, with exposure to different coolants and contaminants, and in different flux fields in order to determine short- and long-term degradations, strengths and weaknesses, potential failure modes, and resistance to shock.

Detailed analysis of performance is then used by the industry to improve designs, codes, methods of manufacturing, material composition, and fuel composition and loading. Regulatory agencies use the data to develop new standards, regulations, and operational parameters. Importantly, test reactors are designed to provide "accelerated data"—a year of data from the test environment may correspond with several years of anticipated operational history. Such data are critical in early resolution of design flaws and in developing modified methods of manufacturing. The data are used by the industry worldwide to drive the innovation necessary for successful deployment of advanced technologies. Without the necessary data, advanced reactors could be unnecessarily plagued by

development and operational problems, mandating in-service repairs, which can be costly and time consuming in the nuclear industry.

Test and operational data also confirm performance of the original design and subsequent modifications, improving online time, eliminating failure modes, and—through increasing robustness—resulting in longer-lived components and potentially decreasing or eliminating the need for servicing and inspection. Such data have a significant impact on economic viability and the extent of regulatory oversight, as well as safety and perceptions of safety both within the nuclear power community and by interested stakeholders.

The different fuel types and coolants impact every aspect of the reactor. Every component within the different neutron fluxes or exposed to differing nonwater coolants required by advanced fuel cycles will have different failure mechanisms. By changing the fuel and coolant types, a whole new set of evaluation data will be required. Reactor components exposed to the higher neutron flux of advanced fuels and differing coolants will be subject to new methods of degradation, which are not currently well understood. For example, a weld on an internal pump or a submerged heat exchanger in a pool reactor may experience different types and speeds of stress corrosion cracking. Additionally, as components come together in the reactor, they will experience different types of thermal expansion and contraction—not only in general, but also based on their position in the reactor. For example, reactor core internals, which will be exposed to both different neutron fluxes and coolants from advanced fuel cycles, will need to be tested extensively to ensure their performance in the new environment. That makes it critically important to have a test reactor with the capability of independent loops that can maintain temperature and pressure (to establish thermal hydraulic testing conditions that will mimic the reactor concept being tested) and simulate specific neutron flux. A test reactor with this capability that supports instrumented leads experiments can help speed up the extensive testing that will be necessary to support advanced fuel cycles.

Over the history of the nuclear industry, there have been a large number of test reactors, which provided the data driving the early industry innovation, improved designs, and development of manufacturing and operating methods and parameters. The early nuclear industry was heavily supported by government policy, and the government directed and funded test facilities, many of which also supported national security policy goals. The early industry was also supported substantially by a strong backbone of dual-use designers and manufacturers that focused on both commercial and national security projects. After a government-sponsored commercial test with the Shippingport demonstration breeder reactor, the nuclear industry settled into tested and established methods, focusing largely on LWRs. As the industry consolidated around LWRs with increasingly established technology—and as facilities aged, computational models matured, and more direct operational data became available—the number of test reactors declined. At the same time, the pursuit of advanced technologies and fuel cycles floundered and generally fell under the purview of government and academic institutions with no realistic path forward toward commercialization.

In the late 1970s, changing economics and the Three Mile Island accident altered the dynamics of the U.S. commercial nuclear industry. While continuing to operate existing plants, the nuclear design, manufacturing, and construction components of the industry significantly contracted, resulting in a severe reduction in the number of nuclear qualified ("N-stamp") firms licensed to perform nuclear work. As the industry contracted, complicated supply chains for Western reactors became more globalized and dispersed, and what had been an almost entirely self-contained and self-reliant U.S. industry now has supply chain partnerships and ownership spanning the globe. With the exception of certain national security programs, this dispersion (and the declining market) has weakened the U.S. nuclear industrial base available for design, engineering, and manufacturing innovation. This weakening in active test reactors, new build expertise, and manufacturing at a commercial level will impact how quickly technology development for advanced reactors and fuel cycles can advance in the United States, especially without considering foreign supply chain expertise.

As mentioned in Section 3.3.2, a major question for development and deployment of test reactors is, Who should pay for the reactors? While Congress is requiring DOE to form public–private partnerships to support the VTR project, DOE officials are concerned that a commercial advanced reactor developer would not want to finance construction of test facilities that would also benefit competitors (Kramer, 2021). Other test reactors at Idaho National Laboratory, such as the Advanced Test Reactor (ATR), have been the government's responsibility for

building and operating. However, one option for funding operations for the VTR or a VTR-like test reactor would be to charge an operational fee for proprietary research but make access free if the experimental results would be made publicly available, while the government would solely or largely pay the construction costs (Kramer, 2021).

The ATR is a one-of-a-kind test reactor. It began operations in 1967, and in 2021 was undergoing its sixth overhaul and upgrades. ATR has provided testing capabilities for U.S. naval reactors, other federal agencies, university researchers, and industry. Although the ATR is a type of pressurized water reactor (PWR), it operates at very low pressures and temperatures compared with commercial PWRs. It uses a beryllium reflector to concentrate neutrons in the core of the reactor in order to provide high-flux thermal neutrons. In addition, ATR has a unique clover-leaf design with lobes that can be operated at different power levels to allow for multiple simultaneous experiments. Industry partners can access the ATR via the National Research Innovation Center, the Gateway for Accelerated Innovation in Nuclear program, or the Nuclear Science User Facilities program (INL, 2021a; NSUF, n.d.). While the ATR has capabilities relevant for testing some types of advanced reactors, the major missing capability is fast neutrons. Researchers at Idaho National Laboratory reported in 2017 that "ATR irradiations performed using cadmium shrouding are sufficiently prototypic that they can be used with confidence in the development and testing of fast reactor fuels" (Harp et al., 2017).

However, the usefulness of such a solution is debatable. Cadmium shrouding does not result in an increase of the fast neutron flux, which means that the ATR is not suitable for shortening the time required for generating and investigating neutron radiation damage in materials used, or planning to be used, in today's or future reactors. Presently, such an accelerated testing capability is only available in the BOR-60, a sodium-cooled fast test reactor that has been operating in Russia for more than 50 years. Direct access to the BOR-60 reactor by U.S. R&D organizations has been difficult at times, and a facility such as the proposed VTR would have significant roles in the development of advanced nuclear energy generation technologies. Worldwide, there is a lack of testing facilities that can produce high-energy neutron fluxes useful for research on fast reactors' fuel and materials (WNA, 2021i).

In the European Union, Belgium has the largest-power (125-MWth) materials testing reactor, the BR2, which has "a unique adaptable core configuration" and has a memorandum of understanding with DOE-NE's Nuclear Science User Facilities to allow U.S.-based researchers to apply to use the reactor (SCK-CEN, 2018). At Cardarache, France, the 100-MWth Jules Horowitz LWR is under construction and is designed for testing of fuels and materials for advanced reactors (CEA, n.d.). This reactor would also be available for international collaboration when completed.

4

Fuel Cycle Development for Advanced Nuclear Reactors

This chapter responds to the first charge of the statement of task calling for an evaluation of the merits and an assessment of the viability of different nuclear fuel cycles, including fuel cycles that may use reprocessing, for both existing and advanced reactor technology options. It builds from the discussions in Chapters 2 and 3 on fuel cycle options for commercial light water reactors (LWRs) and research and development (R&D) programs and needs for advanced reactors, respectively. This chapter addresses the requirements in both the front and back ends of the fuel cycle to support the development and deployment of advanced nuclear reactors. To determine which factors have the greatest impact on the viability of advanced fuel cycles, the chapter examines the proposed merits of advanced fuel cycles and the challenges of achieving those potential benefits, including technical challenges as well as cost and safety aspects.

The chapter begins with a summary that includes the committee's findings and recommendations (Section 4.1). It is then divided into two main sections for the front end (Section 4.2) and back end (Section 4.3) of the fuel cycle. The front-end section examines aspects of mining and milling (Section 4.2.1), conversion (Section 4.2.2), enrichment (Section 4.2.3), and fuel fabrication (Section 4.2.4) relevant to advanced reactor development. In particular, it focuses on the required infrastructure and capabilities for producing and fabricating high-assay low-enriched uranium (HALEU) fuel, which is needed for almost all of the advanced reactor designs examined (Section 4.2.3). Beginning with an introduction to advanced fuel cycles (Section 4.3.1), the back-end section discusses and analyzes reprocessing and recycling options for advanced reactors, under the assumption that plutonium and uranium are considered strategic assets and essential to sustain nuclear power production into the future.[1] It introduces the concept of reprocessing (Section 4.3.2) and then focuses on the recovery and reuse of fissile material, in particular plutonium, from spent nuclear fuel to allow extraction of the maximum energy content possible, analyzing monorecycling of uranium and plutonium in LWRs (Section 4.3.3), multirecycling of plutonium in LWRs (Section 4.3.4), management of minor actinides (Section 4.3.5), and reprocessing methods for advanced reactor fuel (Section 4.3.6). Various motivations, potential benefits, and challenges of implementing advanced fuel cycles (Sections 4.3.8–4.3.11) are considered. Section 4.4 presents the committee's assessment of what is needed for cost estimation for different nuclear fuel cycle options. Finally, Section 4.5 provides information relevant for safety considerations of fuel cycles.

[1] In contrast, Chapter 5 focuses on the direct disposal of spent nuclear fuel as waste.

4.1 CHAPTER 4 SUMMARY, FINDINGS, AND RECOMMENDATIONS

A primary concern for the front end of the fuel cycle for advanced reactors is the production of HALEU fuel. The committee highlights here that the United States will likely not have any significant reliable domestic supply of HALEU for at least one decade—maybe even longer. A sufficient U.S. domestic commercial supply would alleviate the risk of supply reliability, given the great uncertainty over whether the current sole commercial HALEU supplier, Russia, will be allowed to provide HALEU to U.S.-based advanced reactor developers. In response to the Russian war against Ukraine, the U.S. government may impose sanctions against Russian nuclear companies; however, as of March 2022 (as the committee was completing its report), these sanctions had not occurred (see Box 4.1). Additionally, the United States has no commercial-scale facilities for producing other fuel types proposed for advanced reactors (e.g., TRistructural ISOtropic [TRISO] particle, metallic, nitride, mixed oxide, carbide, molten salt liquid, and thorium fuels), although plans for a TRISO fuel fabrication facility are under development. This analysis led the committee to the following finding and recommendation for the front end of the fuel cycle:

Finding 7: There is no current domestic capacity to supply high-assay low-enriched uranium (HALEU) to meet the projected needs of U.S.-based advanced reactor developers over the next decade. Therefore, if reactor projects requiring HALEU continue to advance, identifying a reliable supply of the material will be crucial. Otherwise, many developers will likely initially acquire HALEU from foreign sources, such as Russia, raising concern about ensuring reliable supply. Reliance on foreign sources of HALEU or HALEU feedstock (as many advanced reactor developers had planned to do prior to the invasion of Ukraine by Russia) without a reliable domestic supply could have serious energy and national security implications if advanced reactors using HALEU are adopted widely.

BOX 4.1
Russia and HALEU Supply

In March 2022, as the committee was finalizing its report to enter peer review, Russia's invasion of Ukraine raised concern among some advanced nuclear energy developers and members of Congress about reliance on Russia for nuclear fuel services, especially supply of HALEU. The committee noted that months before the war it recognized the fragility of relying on a sole foreign source for HALEU. The situation is still in flux, with great uncertainty surrounding potential sanctions on Russian uranium and larger disruptions to the nuclear supply chain, but the committee provides here several recent announcements that could impact the development of advanced reactors and fuel cycles in the United States.

Two advanced reactor developers gave public statements to *Wired* magazine in March 2022 that they would not use Russian HALEU for their reactors, despite having previously planned to do so. In particular, Jeff Navin, director of government policy at TerraPower, stated, "We have no interest in supporting a Russian state-owned entity." He explained that while TerraPower had planned to use Russian HALEU to provide the fuel for Natrium's first core, it was anticipating that domestic supplies would be available for subsequent core loadings. Speaking about HALEU from Russia, Jacob DeWitte, the CEO of Oklo, stated, "Frankly, let's be real. I don't think that option's on the table." He acknowledged that "there's a real gap. We can't import it, and we can't produce it yet" (Barber, 2022).

In Congress, Senator John Barrasso (Wyoming), ranking member of the Senate Energy and Natural Resources Committee, introduced a bill in mid-March 2022 that would ban U.S. imports of Russian uranium. His cosponsors included senators Cynthia Lummis (Wyoming), Roger Marshall (Kansas), and Kevin Cramer (North Dakota) (Barrasso, 2022). At the same time, some energy experts warned that an immediate ban could cause a severe spike in international uranium prices. When the Biden administration had raised the potential for such a ban, the uranium spot price shot up to its highest price in more than one decade (Holzman and Northey, 2022). While this situation provides a specific, timely example, it serves the larger purpose of illustrating risks of relying solely on the international market.

Recommendation C: Given the uncertainty of foreign supply arrangements of high-assay low-enriched uranium (HALEU) for advanced reactors, the U.S. Department of Energy should prepare contingency plans that may include (1) scheduled delays in the development, demonstration, and deployment of these systems; (2) a schedule for industry as to when and what level of federal support will be available; and (3) the release of stockpiles of highly enriched uranium for downblending until domestic and secure supplies are available.

On the back end, the advanced fuel cycles described herein represent scenarios where, in the limit, all plutonium and the minor actinides (neptunium, americium, and curium) are recycled and only fission products are left to be disposed of in a geologic repository. Although these advanced fuel cycles are theoretically possible, from a practical point of view, they are costly and challenging to implement as a whole in commercially licensed facilities. These fuel cycles require the construction of complex reprocessing and fuel fabrication facilities, whose reliable, safe, and efficient operation in most cases is well beyond the existing commercial-scale experience base in the United States. While reprocessing options are currently not economically viable in the United States, future events, including decreased availability and increased cost of uranium or major cost escalations of other sources of energy, might nullify these current cost penalties. From that perspective, maintaining modest R&D support for reprocessing technologies would be valuable.

Implementing advanced fast reactors and their associated fuel cycles in order to effectively reduce, but not eliminate (because of inevitable process loss), the quantity of long-lived actinides destined for geologic disposal would have to be operated for many decades to achieve the permanent benefits to the repository and other parts of the nuclear fuel cycle. This necessitates a commitment to nuclear power for several centuries (National Research Council, 1996) and significant financial investment. Implementing a fully closed fuel cycle that includes reprocessing does not eliminate the need for a geologic repository because fission products will still require disposal, material losses will inevitably occur in reprocessing and in other parts of the fuel cycle, and contaminated fuel and process hardware will be generated; therefore, no fuel cycle can be considered as perfectly closed.

Furthermore, for a nuclear fuel cycle supporting any reactor technology to be viable, it has to be *industrially sustainable*. Although different options are available, most would be dramatically different from the current U.S. situation. An industrially sustainable nuclear fuel cycle would require all its elements to be demonstrated individually and together. For that reason, an evolutionary pathway is more practical than a revolutionary approach that attempts to solve all potential issues at the same time.

The committee makes the following findings and recommendations regarding fuel cycles:

Finding 8: For a nuclear fuel cycle supporting any reactor technology to be viable, it has to be *industrially sustainable*. Although many fuel cycle options are possible, most differ dramatically from the current situation in the United States—the once-through fuel cycle. All elements of a sustainable nuclear fuel cycle would have to be fully demonstrated both individually and together, because what works with computer-aided designs would not necessarily translate to industrial-scale deployment. For that reason, an evolutionary, progressive approach is likely more practical than a revolutionary approach that attempts to solve all potential issues at the same time with advanced technologies. The evolutionary approach is more important for commercial deployability and will require the majority of investment efforts; nonetheless, some investments in high-risk, high-reward approaches may be worth pursuing. The committee agrees with the 1996 National Research Council report *Nuclear Wastes: Technologies for Separations and Transmutation*, which states that advanced fuel cycles will require substantial investment and take many decades to more than a century of continuous recycling using a separations and transmutation system of appropriate scale, in order to potentially achieve the full benefit of plutonium recycling and partitioning and transmutation of minor actinides.

Recommendation D: The current U.S. policy of using a once-through fuel cycle with the direct disposal of commercial spent nuclear fuel into a repository should continue for the foreseeable future. The once-through fuel cycle is the baseline, and any new fuel cycles should have advantages over that baseline for them to be deployed. However, so as not to preclude these options in the future, the U.S.

Department of Energy (DOE) should continue fundamental studies to evaluate the feasibility of using recycling and transmutation for closing fuel cycles. Specifically, DOE should develop and implement a phased, long-range research and development program that focuses on advanced separations and transmutations technologies.

Finding 9: As proposed for some advanced reactor closed fuel cycles, reprocessing and recycling of spent nuclear fuel introduces additional safety and environmental considerations over the management of open-cycle light water reactor oxide fuels. In assessing the safety and environmental performance of advanced reactors, the risks and environmental impacts will require optimization over the entire fuel cycle, including front-end processes (mining, enrichment, and fabrication), back-end processes (reprocessing and recycling together), and disposal (interim and final). Currently, advanced reactor developers focus primarily on the safety aspects of the reactor and its operation, and put less priority on the safety aspects of other parts of the fuel cycles.

Recommendation E: Congress and the U.S. Department of Energy should incentivize safety improvements across the supporting fuel cycle.

Finding 10: Because of the absence of current commercial operational experience with advanced reactor technologies in the United States, reliable cost data and estimates for these technologies and their associated fuel cycle components are lacking. The costs of advanced reactors and their associated fuel cycles could range from at least several billion dollars—for pilot-scale non–light water advanced reactors and their fuel cycle facilities—to hundreds of billions of dollars—for full deployment of an alternative fuel cycle that would replace the existing once-through cycle and existing light water reactors. Congress and the U.S. Department of Energy will need better understanding of the cost estimates for various scenarios of reactor deployment and supporting fuel cycle requirements to aid their decision making as to what technologies to support in the coming years.

Recommendation F: Congress and the U.S. Department of Energy should obtain an independent assessment of cost estimates of various scenarios for potential deployment of advanced reactor technologies and related fuel cycle components. The independent assessor should have expertise in evaluating large-scale construction projects; examining project management challenges; and understanding technological and financial risks, as well as their uncertainties.

4.2 FRONT END OF THE FUEL CYCLE TO SUPPORT FUEL PRODUCTION FOR ADVANCED NUCLEAR REACTORS

The objective of the front end of the fuel cycle is to make fuel for reactors. Chapter 2 covers the steps of the front end relevant for the once-through and monorecycling fuel cycles for LWRs. This section focuses on aspects of the front end relevant for non-LWR advanced reactors, as defined in Chapter 3. One major aspect is the use of HALEU, which is a requirement for fueling most of the advanced reactor designs under development (see Table 3.1).

In this report, HALEU is defined as enrichment of uranium-235 greater than or equal to 10 percent but less than 20 percent. The commercial LWR fleet currently uses low-enriched uranium (LEU) fuel with enrichments between 3 and 5 percent uranium-235. Enrichments from 5 percent to just under 10 percent are considered LEU+. While the interdependent, international market for the front-end services has worked effectively to supply U.S. LWR fuel needs, developers have expressed concern to the U.S. Department of Energy (DOE) about whether the same will be true for advanced nuclear reactor fueling requirements—particularly whether HALEU will be available (Caponiti, 2020a).

Advanced reactors use HALEU fuels for different reasons. Some small and microreactor designers plan to use HALEU because the higher enrichment is needed to counteract neutron leakage from the smaller reactor cores. Also, fast reactors cannot achieve criticality with lower-enriched fuel. In contrast, thermal reactors such as high-temperature gas-cooled reactors (HTGRs) do not require HALEU, but some designers choose to use HALEU to allow the fuel to reach higher burnups, which can improve fuel utilization, or to reduce the frequency of refueling.

However, use of HALEU fuel generally reduces natural uranium utilization efficiency because of the larger amount of mined uranium needed to generate the HALEU, unless higher burnups are achieved.

The following subsections provide information on U.S. and international capacities for the specific steps of the front end: mining and milling, conversion, enrichment, and fuel fabrication, with particular emphasis on addressing the challenges of supplying HALEU.

4.2.1 Mining and Milling

As introduced in Chapter 2, the mining and milling of uranium provide the raw uranium material required for nuclear fuel. Concerns about increasing U.S. dependency on uranium imports from foreign state–owned enterprises (e.g., those in Russia, China, and Kazakhstan), led two U.S. domestic uranium mining companies to petition the U.S. Department of Commerce (DOC) in January 2018 to investigate the effects of state-subsidized foreign uranium supplies on the domestic mining industry and the potential impacts of these imports on national security (Energy Fuels, 2018). In May 2019, DOC reported to President Donald J. Trump that these imports posed a threat to national security and recommended action under Section 232 of the Trade Expansion Act of 1962 (19 U.S.C. § 1862), which gives the president the authority to restrict certain imports if DOC determines that these "impair national security" (Larson, 2020).

While President Trump did not concur with DOC's findings, the administration expressed significant concerns and responded by creating the Nuclear Fuel Working Group (NFWG), which was assigned to examine options for reviving domestic nuclear fuel production, as well as the entire nuclear supply chain. In early 2020, the Trump administration proposed $150 million in its fiscal year (FY) 2021 budget to Congress to build a strategic uranium reserve that would (1) provide a supply of domestic uranium to commercial nuclear power production in the event of an international market disruption and (2) ensure sufficient domestic uranium capacity for U.S. defense needs. However, according to the Congressional Research Service, DOE did "not identify what would constitute a market disruption, criteria for providing uranium material to commercial utilities, or the price at which the uranium from the stockpile would be sold" (Larson, 2020). Using the then-current uranium spot price of $24.90 per pound milled uranium oxide (U_3O_8), the requested $150 million could purchase about 2.7 million kilograms of U_3O_8 from domestic suppliers (Larson, 2020), which would only meet about 13 percent of the annual demand from U.S. nuclear power plants. At the time, $75 million was provided (WNN, 2020a).

Protections sought by U.S. uranium mining companies and Urenco, the only active commercial uranium enrichment company in the United States, were realized with the October 2020 agreement between DOC and Rosatom, the Russian state nuclear energy corporation, to extend the suspension of the U.S. antidumping investigation until 2040 (NEI, 2020a). This agreement allows Russia to continue exporting enriched uranium to the United States, but it reduces the proportions from approximately 20 percent of U.S. annual demand to no higher than 15 percent from 2028 to 2040. The amendment also limits the natural uranium and uranium conversion services from Russia to an amount equivalent to no more than 5 percent of U.S. annual enrichment demand from 2026 to 2040. This agreement is relevant for meeting the fuel requirements for advanced reactors because it provides "predictable rules" and would not significantly impact Russia's ability to supply fuel services of the market for existing U.S. reactors and the potential market for U.S. advanced reactors, according to Rosatom after the agreement was concluded (WNN, 2020b). However, because of the Russian war against Ukraine, the U.S. government may impose sanctions on imports of Russian uranium and nuclear fuel services. For additional information on what was known as of March 2022 as the committee was completing its report, see Section 4.2.3.3 and Box 4.1.

4.2.2 Conversion

As described in Chapter 2, milled uranium oxide (U_3O_8) is converted to uranium hexafluoride (UF_6) prior to enrichment. The October 2020 U.S.–Russian amended agreement on antidumping of Russian uranium also affected the business decision of Honeywell-ConverDyn, the sole U.S. uranium conversion company, about restarting its conversion plant. In 2018, the company had stopped production at its Honeywell Metropolis Works (MTW) conversion plant in Illinois (capacity of 7,000 MT [metric tons] uranium per year) because of low demand and high UF_6 inventories

globally. In early 2021, Honeywell announced plans for the MTW to restart in 2023 after completing plant upgrades. Honeywell chief executive officer (CEO) Malcolm Critchley stated in a press interview that before spending the $150 million required for the plant's upgrades, he and his team needed to assess the business case, which involved several pieces. One was assurance that the Russian conversion capacity would be kept at a relatively low level for supplying the U.S. reactor fleet. Second was the continuation of the NFWG under the Biden administration, indicating bipartisan support for reviving domestic nuclear fuel production. Third, in 2020, the company's customers expressed strong interest in long-term contracts, thereby helping to lower the financial risk of restart. Finally, in 2020, the U.S. Nuclear Regulatory Commission (U.S. NRC) approved the license extension of MTW for 40 years—in the past, the license extensions had been valid for only 10 years. This license approval further reduced financial uncertainty around reopening the MTW (WNN, 2021a). Having a U.S. uranium conversion company in long-term operation is an important step toward a domestic nuclear fuel supply chain. However, its benefit to U.S. advanced reactor developers would be limited unless domestic HALEU enrichment capacity can meet demand, a challenge discussed in the next section.

4.2.3 Enrichment, Production, and Supply of HALEU

Establishing commercial-scale production and supply of HALEU will be an essential enabling factor in the deployment of advanced reactors, since, as noted above, most advanced reactors under development require HALEU for their fuel. The U.S. government has ongoing efforts to develop HALEU production capability through enrichment of natural uranium, downblending of highly enriched uranium excess to nuclear weapons, and reprocessing and downblending of highly enriched uranium spent nuclear fuel from research and test reactors. Production of HALEU via enrichment of natural uranium is also being explored by private industry.

Congress recognized the potential need for HALEU in the Energy Act of 2020, which authorized DOE to establish a HALEU Availability Program to "support the availability of HALEU for civilian domestic research, development, demonstration, and commercial use" (DOE-NE, 2021e). DOE released a request for information in the form of an extensive questionnaire regarding this program in December 2021, with an extended due date of February 14, 2022, for receipt of information (DOE-NE, 2021e). Despite these efforts, significant timing, cost, and regulatory challenges remain for building a robust domestic HALEU supply chain. This section examines the status and challenges of HALEU production and supply, provides discussion about ways to address production and supply, and concludes with information on safety and security requirements for HALEU.

4.2.3.1 Domestic HALEU Production Capabilities

The sole operating commercial enrichment facility in the United States, the Urenco plant in New Mexico, is currently not ready to produce HALEU. Urenco, a foreign-owned company, claims that its New Mexico plant has scope for expansion to accommodate a facility meeting Category II security requirements (see Sidebar 4.1) to produce HALEU and that "if detailed design, site permits, and contractor selection were undertaken in parallel with the regulatory licensing process," it could have a HALEU production unit ready within 24 months of regulatory licensing approval (Fletcher, 2020).

DOE is working to stimulate the development of a domestic U.S.-based HALEU production capability, most notably through a 3-year, $170 million cost-shared demonstration project with Centrus Energy Corporation to show the viability of advanced centrifuges for making HALEU. As of 2021, Centrus had built 16 of these centrifuges with the aim of producing up to 600 kg of HALEU by June 2022. In June 2021, the U.S. NRC approved Centrus's request to amend its license to produce HALEU up to 20 percent enrichment (DOE-NE, 2021f). In November 2021, Centrus announced that supply chain difficulties due to the COVID-19 pandemic have affected the production schedule and stated that production would not begin until mid-2022 (Patel, 2021). Beyond the June 1, 2022, end date for Centrus' current contract with DOE, the company could receive additional funding through DOE's proposed HALEU Availability Program, to be started in FY2022, although this would require them obtaining a new, competitively awarded contract (Patel, 2021). However, several members of the House of Representatives have expressed concern that the competitive bidding process will not be truly competitive (Kramer, 2022). In particular, the presolicitation announcement on February 7, 2022, by DOE's Office of Nuclear Energy (DOE-NE) limits the

bids to those who can produce HALEU at the 16-centrifuge facility and specifically the "place of performance will be at the DOE-owned American Centrifuge Plant in Piketon, Ohio" (DOE, 2022d).

An alternative HALEU production method is downblending, the process by which higher enrichment material is mixed with lower-enrichment material to obtain the desired specified level of enrichment. The United States has declared 374 MT of highly enriched uranium as excess to nuclear weapons; 152 MT have been set aside to fuel U.S. naval reactors on submarines and aircraft carriers; and 28 MT were made available to have U.S.-origin material to fuel the Watts Bar Nuclear Power Plant, which has the mission of producing tritium, a hydrogen radioisotope used in nuclear weapons. In addition, the United States has been using the remaining part of the declared excess highly enriched uranium to meet the needs of fueling research reactors and isotope-production reactors (DOE, 2015b). Because the enrichment levels of the highly enriched uranium stocks are classified, the committee was not able to calculate how much HALEU could be produced from downblending available highly enriched uranium stocks.

Currently, two facilities have conducted downblending activities: BWXT's Nuclear Fuel Services (NFS) facility in Erwin, Tennessee, a designated Category I facility licensed by the U.S. NRC to handle highly enriched uranium, and the DOE's Y-12 facility at Oak Ridge, Tennessee. The NFS is the only U.S. commercial company licensed and capable of downblending MT quantities of highly enriched uranium, but as of April 2020, no funding was available for this downblending, although the company has been able to show production of demonstration quantities of HALEU (e.g., for TRISO fuel) (Nagley, 2020). BWXT claims that, with adequate funding, NFS would be prepared to add equipment to its existing capacity and could downblend 1–2 MT of highly enriched uranium annually to produce up to 10 MT of HALEU. In addition, BWXT has demonstrated deconversion capabilities to turn the downblended uranium compounds into oxides, metals, and various other compounds including nitrides and silicides. Deconversion on-site would avoid shipping HALEU as UF_6 (Nagley, 2020). In a January 2021 briefing to the committee, BWXT mentioned that the company has developed a near-term strategy to use U.S.-sourced natural uranium feedstock for producing about 3 MT/year of unobligated HALEU that would be free of peaceful-use obligations (which is not a requirement for most power-related uses of HALEU). BWXT estimates it will take 5–6 years to construct, commission, and begin production at the proposed new facility (Nygaard, 2021).

Another method for having a near-term U.S. capacity for HALEU is the use of reprocessed uranium, though there are some significant technical challenges with this option. Reprocessed uranium can come from research and test reactors, as well as isotope-production and naval reactors fueled with highly enriched uranium. For example, the Savannah River Site has capability for reprocessing at its H-Canyon facility and could meet some needs for HALEU (Bates, 2020). From 2003 to 2011, the Savannah River Site produced 301 MT of 4.95 percent–enriched uranium for commercial reactors at the Tennessee Valley Authority. The recycling operations have been performed on irradiated fuels from research, test, and isotope production reactors. The Savannah River Site is storing highly enriched uranium solution that could be converted into about 2 MT of 19.75 percent HALEU by late FY2022, and is awaiting a decision about what to do with that material. In addition, the site could produce even greater amounts of HALEU: in particular, up to 19 MT from reprocessing DOE's aluminum-based spent nuclear fuel, which is stored at the site, and up to 13 MT from reprocessing Idaho National Laboratory's spent nuclear fuel. Production of 1 to 1.5 MT per year could start in FY2023 (Bates, 2020).

However, the Defense Nuclear Facilities Safety Board has documented safety issues at H-Canyon that would have to be addressed to allow for extending operations at this facility (DNFSB, 2021). In April 2022, DOE decided to not pursue recovery of highly enriched uranium from 29.2 tonnes of heavy metal of research-reactor spent fuel currently in storage at the Savannah River Site for downblending to HALEU, but it instead will dissolve the spent fuel (containing highly enriched uranium) and immobilize it into borosilicate glass for storage until a repository is available (DOE, 2022e).

Seeking additional options for HALEU fuel production, Idaho National Laboratory (INL) undertook a study in 2019 to evaluate the technical and cost requirements for establishing a metallic or ceramic/intermetallic-type (e.g., pellet) HALEU fuel fabrication line in each of three existing buildings on-site, using as feedstock sodium-bonded metallic highly-enriched uranium fuel that was irradiated in Experimental Breeder Reactor-II (EBR-II) and electrometallurgically treated (Crawford et al., 2019). INL's planning assumed a 2.5-MT throughput,[2] and because the

[2] The actual production rate for a typical commercial fuel fabrication facility is 1–2 orders of magnitude higher than INL's planning assumption.

starting material was known to contain residual contamination from both transuranic elements and fission products, use of inert gas–filled glove box enclosures was required. The rough order-of-magnitude cost estimate for building modifications started at $10 million and increased commensurately with other activities depending on the specific building, such as relocation of existing programs, decontamination, and decommissioning. The cost estimate for fabrication equipment and glove boxes totaled $22.5 million for the metallic fuel line and $28 million for the ceramic/intermetallic fuel fabrication line, with uncertainties of –20 to +50 percent. The annual operating cost for a fuel fabrication line was estimated to be around $7 million in 2019 dollars. Also important to note, industry feedback to DOE-NE indicated that HALEU derived from EBR-II spent fuel could not be used for TRISO fabrication because "(1) The residual impurities and radioactive contamination impose the requirement for excessive transportation, handling, shielding, and protective equipment that will significantly add to the fuel fabricator's manufacturing costs. (2) The EBR-II HALEU, produced at the INL, cannot leave the site, so fuel fabrication would have to be performed there. There are no current plans for TRISO based fuel manufacturing at INL" (Caponiti, 2020a).

To satisfy defense-related needs for enriched uranium, in 2017, DOE's National Nuclear Security Administration (NNSA) initiated efforts to establish a domestic uranium-enrichment capability to address the need for a reliable supply of unobligated LEU at 19.75 percent enrichment. Three specific needs were called out: (1) HALEU for research, test, and demonstration reactors by 2030; (2) low-enrichment uranium reactor fuel for tritium production by 2038; and (3) highly enriched uranium for naval reactor fuel by 2060. A 2018 assessment of NNSA's plan by the U.S. Government Accountability Office (GAO) recommended

1. "The NNSA Administrator should revise the scope of the mission need statement to clarify which mission need it seeks to achieve and, as appropriate, adjust the range of options considered in the analysis of alternatives process.
2. The NNSA Administrator should—following clarification of the scope of the mission need statement—ensure that the agency's cost estimates for whichever options it considers going forward are aligned with the scope of the mission need that the enrichment capability is intended to fulfill and that they are developed consistent with best practices" (GAO, 2018).

However, NNSA disagreed with these recommendations.

4.2.3.2 Challenges and Opportunities for Establishing Domestic HALEU Supply

Major steps that would be needed to produce HALEU fuel for advanced reactors were outlined at INL's Gateway for Accelerated Innovation in Nuclear workshop on HALEU in April 2020:

- Develop, on concurrent schedules, the enrichment, deconversion, and fabrication facilities.
- Make sure the regulatory resources are available to support the licensing framework for the HALEU fuel cycle.
- Match the timing of the front-end fuel cycle development for next-generation fuels with the forecasted aggregate demand from the advanced reactor vendors.
- Provide sufficient assurance of a reasonable and necessary economic return for companies investing in HALEU facilities (Fletcher, 2020).

Industry feedback to DOE-NE, shared at the same INL workshop, reflected these challenges and proposed some potential solutions. In particular, multiple industry representatives urged DOE to support the entire infrastructure for fuel production for advanced reactors, including enrichment, fabrication (of various fuel types), transportation, and licensing (Caponiti, 2020a). Industry representatives also stressed the importance of HALEU being commercially available at an economically competitive price in a timely fashion (within 3–4 years). The industry's recommended actions for DOE included guaranteeing an interim supply of HALEU for demonstration reactors, assessing future procurement of HALEU for a fuel bank, establishing fuel fabrication facilities to handle HALEU and accommodate multiple fuel types, and licensing a HALEU shipping container (Caponiti, 2020b).

During a briefing to the committee in June 2021, Daniel Poneman, CEO of Centrus, indicated that the primary dilemma for HALEU supply is somewhat of a "chicken and egg" problem: who will buy U.S. advanced reactors if the United States lacks a guaranteed fuel supply—but who will invest in the fuel supply infrastructure without a guaranteed customer base? He put forward the model of leveraging U.S. government demand for HALEU in order to build domestic capacity to meet national security needs, as well as needs for advanced nuclear reactor development, noting that "DOE should lead the way" (Poneman and Cutlip, 2021). Another challenge, however, comes from the different economies of scale for low-enriched uranium versus HALEU: there is a known and reliable demand of about 15 million SWU/year (separative work units per year) for low-enriched uranium for the LWR fleet, but there is an uncertain future demand for HALEU, and the government's near-term need is less than 100,000 SWU/year. In addition, as a Category II nuclear material (see Sidebar 4.1), HALEU has additional security costs. To help reduce this cost and risk, Centrus plans to collocate enrichment and deconversion on one site (Poneman and Cutlip, 2021). Similarly, Urenco plans to collocate enrichment and deconversion for security reasons (Fletcher, 2020).

Recognizing the challenges and concerns regarding HALEU availability, DOE-NE plans to continue its coordination with other agencies and partnerships with industry to make progress on HALEU production in FY2022 (Griffith, 2021). In particular, DOE-NE will work with NNSA on the recovery and downblending of limited quantities of highly enriched uranium to HALEU. It will also "develop criticality benchmark data to support the design and licensing of transportation packages" and "acquire transportation packages for Department of Energy-owned HALEU" (Griffith, 2021). Furthermore, DOE will continue support for the HALEU-enrichment demonstration facility, sharing costs with industry, and will provide additional support to this demonstration contractor "for impacts related to COVID, such as supplier issues and contractual impacts" (Griffith, 2021). DOE-NE will also work "with industry to understand and help enable commercialization of long-term private-sector HALEU production" (Griffith, 2021); however, it does not intend to be the sole funder of HALEU production and thus requires private-industry investment. Finally, DOE-NE plans to "initiate National Environmental Policy Act activities supporting HALEU availability" (Griffith, 2021). As this report was being finalized, DOE-NE was receiving responses to its request for information in the *Federal Register* (DOE-NE, 2021g).

4.2.3.3 Foreign HALEU Supply

The one currently available foreign supplier of HALEU is Russia's Rosatom and its company TENEX, which supplies uranium products from the Russian Nuclear Fuel Complex (NFC). This complex includes the Novosibirsk Chemical Concentrates Plant (fuel and targets fabrication), the Siberian Group of Chemical Enterprises (conversion services), and the Production Association Electrochemical Plant (enrichment and deconversion services). The Russian NFC has many years of experience producing relatively small quantities of HALEU for research reactors and has certified shipping containers for this material. It would need to have higher-capacity certified containers for commercial-reactor quantities of HALEU, but TENEX officials believe that the experience with the existing TUK-159 containers (35 kg uranium oxide or up to 50 kg uranium metal) could form the basis for developing higher-capacity containers. At INL's April 2020 HALEU Workshop, these officials stated that the Russian NFC could produce HALEU in metal or oxide forms within 6–9 months after receipt of an order (Newton and Kolosovskaya, 2020). See Box 4.1 for the political and legal situation as of March 2022 regarding potential U.S. sanctions on importation of Russian nuclear fuel services.

Potentially, Orano could also produce and supply HALEU, though its enrichment facilities in France are not currently configured for this purpose. During the committee's September 2021 information-gathering meeting, Amir Vexler, CEO of Orano USA, acknowledged that Russia is presently the only commercial supplier of HALEU and said that "Orano is committed to fueling the future of nuclear energy through the development of commercial platforms that secure HALEU production capacity and the associated logistics infrastructure." But the diversity of reactor operations' enrichment and deconversion requirements will need "a flexible platform for HALEU production," leveraging "integrated advanced chemistry applications backed by Orano's expertise and delivered through industrial partnership." Orano is also closely monitoring demand signals from the United States and needs "greater clarity" about commercial commitments and "the availability of policy tools that can accelerate investment in this critical infrastructure" (Vexler, 2021).

If advanced non-LWR developers deploy demonstration reactors in the next decade, they will need amounts of HALEU that current U.S. sources might not be able to supply. As stated above, currently the only source of commercial HALEU is Russia, where legal and political barriers are in flux (see Box 4.1). In his remarks to the committee, Daniel Poneman of Centrus Energy expressed that U.S. utilities would want at least one assured domestic source before they made a commitment on a HALEU-fueled reactor for 30–60 years (Poneman and Cutlip, 2021). In other words, foreign imports can provide price competition, but not fuel assurance. Furthermore, although downblended or reprocessed highly enriched uranium can provide some limited amounts of HALEU, this source is insufficient for fuel assurances spanning the operational life of utilities' advanced reactors (Poneman and Cutlip, 2021).

4.2.3.4 Safety and Security Requirements for HALEU

In developing a HALEU supply chain to support advanced reactors, safety and security considerations must be addressed. First and foremost, enrichment and fuel fabrication facilities will have to conform to radiological and safety standards. HALEU will require Category II special nuclear material facilities (see Sidebar 4.1 and Chapter 6). Other issues include having available U.S.-certified shipping containers for enrichments between 10 and 19.75 percent (for UF_6, uranium oxide, and uranium metal, and as manufactured fuel forms); criticality benchmarks for licensing facilities and transport packages; and updated regulatory requirements and guidance for nuclear material control and accounting, as well as physical security, that address issues specific to HALEU. These transportation and regulatory issues are discussed in greater detail in Chapter 6.

Use of higher enrichments in HALEU means that margins to inadvertent criticality are reduced. Criticality accidents could occur if there is inadequate regulatory oversight, lack of an appropriate safety culture, or deficient worker training and qualification (U.S. NRC, 2000). Such problems have occurred; for example, in 1999 at the JCO Fuel Fabrication Plant in Japan, an inadvertent criticality event occurred when operators attempted to process uranium fuel at 18.8 percent enrichment using a technique they had successfully used previously to process uranium with 6 percent enrichment.

Fuel fabrication facilities that process HALEU will have other significant requirements beyond criticality control associated with whether the final form of the fuel is metallic, ceramic (e.g., UO_2, UN, UC), intermetallic (e.g., U_3Si_2), or alloy (e.g., UMo, UZr). These might include fire protection for pyrophric materials and additional chemical safety and radiological controls depending on the specific process being used. Chemical, radiological, and criticality safety standards are likewise important for on-site storage and transportation of HALEU-containing materials at fuel fabrication facilities. All of these changes will also be accompanied by another level of nuclear material control and accounting and physical security requirements because of HALEU. These requirements may prohibit current Category III facilities from reasonably transitioning to a Category II facility without massive modifications and retrofits or substantial redesign of the existing facility. (See Sidebar 4.1 for category definitions.) It might be most cost effective to design, construct, and license a new Category II facility for fuel fabrication rather than modifying an existing Category III facility to a Category II facility and amending the license. (See Chapter 6 for more discussion about security risks.)

4.2.4 Fuel Fabrication

Domestic fuel fabrication is the only piece of the uranium supply chain that is currently sufficient to meet the needs of the U.S. commercial power industry for uranium oxide fuel with less than 5 percent enrichment. However, the existing domestic fuel fabrication facilities are not equipped to handle HALEU or to produce non–uranium dioxide fuels. Thus, fuel fabrication capabilities will have to be built to meet the needs of advanced reactors requiring HALEU and/or using fuel types other than uranium dioxide. A primary challenge for starting an industry for advanced fuel production, as identified by the Nuclear Energy Institute, is that the reactor designers and fuel producers cannot proceed early in the process unless each side is certain that the other will reach commercial deployment (Nuclear Energy Institute, 2018). Other challenges include the lengthy time required to

SIDEBAR 4.1

Special nuclear material (SNM) is separated into three categories.

- Category I (strategic SNM): 2 or more kg uranium-233 or plutonium; or 5 kg or more uranium-235 contained in uranium enriched to 20 percent or more.
- Category II (SNM of moderate strategic significance): 10 kg or more of uranium-235 enriched to 10 percent or more but less than 20 percent uranium-235.
- Category III (SNM of low strategic significance): 10 kg or more of uranium-235 contained in uranium enriched above natural but less than 10 percent uranium-235.

SOURCE: U.S. NRC, 2020l.

garner commercial support for funding, to address regulatory issues involving both the reactor designs and the fuel enrichments, and to build up the overall fuel cycle infrastructure (Nuclear Energy Institute, 2018).

As mentioned in Chapter 2, three Category III fuel fabrication plants are currently licensed by U.S. NRC and fabricate low-enriched uranium fuel that is sold worldwide to the LWR community: Global Nuclear Fuel-Americas in Wilmington, North Carolina; Westinghouse Columbia Fuel Fabrication Facility in Columbia, South Carolina; and Framatome, Inc., in Richland, Washington. In addition, as noted in Section 4.2.3, two U.S. facilities are licensed to fabricate highly enriched uranium fuel primarily for defense applications: Nuclear Fuel Services in Erwin, Tennessee, and BWXT Nuclear Operations Group in Lynchburg, Virginia. Both are designated as Category I facilities. They have produced both high- and low-enriched uranium fuel for the naval reactors program as well as greater than 5 percent–enriched fuel for nonpower reactors (test, medical isotopes, and research [and training] reactors) by downblending highly enriched uranium. These facilities, in principle, could produce HALEU fuels with modest amendments to their licenses; however, having commercial and defense-related enrichment activities at the same location is not desirable, and the United States has long avoided mixing defense and commercial nuclear activities so as not to encourage this behavior in other countries.

The fabrication of new fuels for advanced reactors will likely require different processes than those described in Chapter 2 for fabricating uranium dioxide fuel assemblies for the existing LWR fleet. New fuels, most of which are based on HALEU as the starting material,[3] are being developed to complement advanced reactor designs that are intended to perform more efficiently at high temperatures and to be more accident tolerant under off-normal conditions. The remaining parts of this section focus on fabrication of non-LWR fuels. Because several advanced reactor developers want to use TRISO-type fuels, including X-energy, which received a major award under DOE-NE's Advanced Reactor Demonstration Program (ARDP), Section 4.2.4.1 has an extensive discussion of their fabrication. Similarly, Section 4.2.4.2 includes substantial coverage of metallic fuel fabrication because of the interest in this fuel type proposed by TerraPower's Natrium, also a major-award recipient under the ARDP. Section 4.2.4.2 also provides information relevant for understanding fuel fabrication challenges of the other advanced reactor fuel types, including nitride, mixed oxide, carbide, and molten salt liquid fuels.

4.2.4.1 TRISO Fuel Development and Production

TRISO-coated particle fuel is at the heart of many high-temperature reactor designs. These designs include high-temperature gas-cooled reactors (HTGRs), such as X-energy's Xe-100 helium-cooled pebble-bed high-

[3] The enrichment limitation of 5 percent for LWRs is also under study at the request of Member States of the International Atomic Energy Agency, who want to realize potential economic benefits from higher fuel burnup, longer fuel cycle operation, and reduced used fuel inventory (IAEA, 2020d).

temperature reactor and Framatome's Steam Cycle High Temperature Gas-Cooled Reactor, as well as fluoride salt–cooled high-temperature reactors (FHRs), such as Kairos' XP-X fluoride-cooled [FLiBe][4] pebble-bed high-temperature reactor.

TRISO fuel was first developed in the late 1950s to support the Dragon Reactor in the United Kingdom. The initial process involved applying a pyrocarbon[5] layer to protect the uranium carbide fuel kernels during fabrication, but this technique rapidly evolved to the application of many layers to prevent the loss of fission products from the kernel (Demkowicz et al., 2019). Current TRISO fuel particle designs use a spherical kernel with a diameter of 350–500 microns, which contains the nuclear fuel (e.g., uranium dioxide or UCO, a mixture of uranium oxide and uranium carbide), which is then surrounded by three layers of carbon and one layer of silicon carbide (SiC). The first layer is a porous carbon buffer that provides space for fission gases to migrate and accommodates kernel swelling. The next is a pyrolytic carbon layer that acts as a barrier for diffusion of fission products and serves as the mechanical substrate for the SiC layer. The SiC layer functions both as the primary barrier for nongaseous fission products and as the structural, load-bearing component of the fuel particle. The fourth and final layer is made of pyrolytic carbon that further aids in the retention of fission gases, protects the underlying SiC layer during handling, and provides a surface for bonding to the fuel-form graphite matrix. After the graphite is added, the mixture is pressed into its final shape (pebbles or prismatic compacts). For pebbles, pressing is done in two stages: the first pressing is of the fuel center, where the coated particle fuel kernels are evenly distributed throughout the volume. The second pressing then produces an outer fuel-free zone of ~5 mm (Mulder, 2021; Nygaard, 2021; Pappano, 2021).

The fuel types within the kernels of TRISO particles can vary widely depending on the reactor design. The fuel types can be either fissile (e.g., UC_2, PuO_2, $(Th,U)C_2$, $(Th,U)O_2$, UO_2, UCO) or fertile (e.g., ThC_2, ThO_2, UO_2, UCO) materials. The fission reaction using UCO as the fuel produces less carbon monoxide (a product of irradiation-induced chemical reactions), thereby generating less internal gas pressure in the pyrolytic layers of TRISO fuel particles under reactor operating conditions. As a result, UCO has shown superior fuel performance at high burnup. For this reason, many high-temperature reactor developers are choosing UCO as the reference fuel type for their TRISO particles (EPRI, 2019), typically in one of two fuel configurations (see Figure 4.1). In the first configuration, TRISO particles are fabricated into cylindrical pellets or compacts and embedded in channels in prismatic graphite blocks. In the second, TRISO fuel particles are dispersed in a graphite matrix and formed into spheres or pebbles about the size of a billiard ball.

Their structural resistance to neutron irradiation, corrosion, and oxidation makes TRISO fuels very robust. Furthermore, because they do not melt under extreme temperatures, such as those found in loss-of-coolant scenarios, TRISO fuels are considered accident tolerant. TRISO developers also claim the added benefit that, because of the layering process, fission products are retained in the TRISO particles; thus, the fuel form acts as its own containment vessel. Although the layered approach used in making TRISO fuel offers a high degree of containment for fission products, the containment is not perfect, and there is limited operating experience at commercial scale using this fuel. Additional information on TRISO fuel, particularly with respect to its storage and disposal as waste, can be found in Section 5.5.2 in Chapter 5 and Appendix G.

Two companies, BWXT and X-energy, are taking steps to manufacture TRISO fuels in the United States. BWXT announced in November 2020 that its TRISO nuclear fuel line project is actively producing fuel at its Lynchburg, Virginia, facility with the intent of meeting application needs for the Department of Defense and demonstration needs for the National Aeronautics and Space Administration (ANS, 2020). Plans for this facility include scaling up TRISO production with full scrap recovery to ~1 MT per year, if needed, by downblending highly enriched uranium to HALEU in the short-term. BWXT is actively considering adding additional TRISO manufacturing capacity in 1 MT/year modules at another of its existing U.S. NRC–licensed facilities.

X-energy is currently making TRISO-coated particles at its TRISO-X Pilot Facility located at Oak Ridge National Laboratory (ORNL); this facility was developed using private and government funding through DOE's

[4] FLiBe = 2^7LiF-BeF_2.

[5] Pyrocarbon, or pyrolytic carbon, is a robust, graphite-like material deposited from gaseous hydrocarbon compounds on suitable underlying substrates (carbon materials, metals, ceramics) at high temperatures in the absence of oxygen.

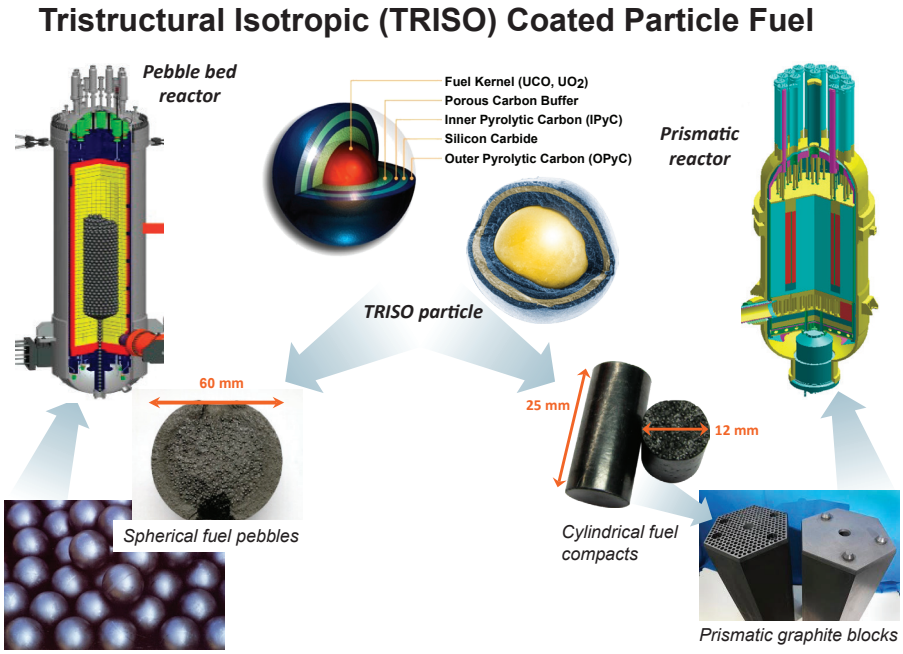


FIGURE 4.1 Reactor configuration using TRISO fuel particles as pebbles in a pebble-bed reactor or a cylindrical fuel compact embedded in graphite blocks in a prismatic reactor.
SOURCE: Adapted from Demkowicz (2019), slide 10.

Advanced Reactor Concepts Cooperative Agreement, awarded to X-energy in 2016. The Xe-100 fuel form is a pebble of 60 mm diameter, weighing ~200 g, and containing ~1 g uranium-235 with dilution of uranium throughout for proliferation resistance. The fuel core of the TRISO pebble contains roughly 19,000 TRISO particles uniformly throughout, with an exterior 5-mm-thick fuel-free zone.

Operational since 2018, X-energy's TRISO-X Pilot Facility was developed through a public–private partnership. ORNL supplies 5,000 sq. ft. of processing floor space, but the processing equipment is owned by X-energy. Most of the work to date has been processing with depleted and natural uranium, but the processing area is authorized to handle research quantities of HALEU to support fuel design, manufacturing, and licensing for the Xe-100 reactor. The facility is referred to as a "pilot facility" because only one fuel fabrication production line is being developed and tested under the American Society of Mechanical Engineers' Nuclear Qualification Assurance Standard. However, all of the equipment in the pilot facility has been developed to be used at commercial scale, which has the potential to significantly reduce risks associated with fuel qualification and simplify the process of scaling production capacity in a commercial setting by simply replicating the existing equipment and increasing the number of production lines in a HALEU-licensed facility.

In November 2021, X-energy announced that its preliminary design for its fuel fabrication facility was completed (WNN, 2021c). X-energy believes this facility will be the first Category II licensed facility capable of handling HALEU of up to 19.75 percent enrichment. X-energy plans to obtain the needed HALEU feedstock from commercial sources. The company anticipates the complete U.S. NRC licensing review to take 36 months, and the facility is scheduled to be operational in early 2025 to meet the Xe-100 schedule for deployment in 2026–2027. In April 2022, X-energy selected the Horizon Center Industrial Park in Oak Ridge, Tennessee, as the location of its first commercial TRISO-X fuel fabrication facility (X-energy, 2022). X-energy states that the ARDP funds all elements of the Xe-100 plant deployment, including the commercial deployment of the TRISO-X Fuel Fabrication Facility (TF3): license application submittal, U.S. NRC license review, fuel processing and material-handling equipment procurement, setup, operational readiness reviews, and manufacturing a sufficient number of fuel pebbles to meet the first core load requirements of the Xe-100 reactor plant (Pappano, 2021).

4.2.4.2 Metallic Fuel Development and Production

Although metallic fuels are not currently in use in power reactors, they are being researched actively in Russia, the United States, China, and Japan. The United States has previous experience with these fuels—for example, in the operation of EBR-I, EBR-II, the Fast Flux Test Facility, and Fermi-1. TerraPower plans to use sodium-bonded metallic fuel in its Natrium demonstration reactor, slated for initial operation by 2027, and to eventually move to a metallic fuel form with no sodium bond (Hejzlar, 2021; Neider, 2021). TerraPower is conducting irradiation testing of its advanced fuel and, to date, has demonstrated good behavior to over 180 GWd/MT (TerraPower, 2022). ARC Clean Energy and Oklo also plan to use metallic fuel in their ARC-100 and Aurora reactors, respectively. Lightbridge Corporation is developing metallic fuel for use in LWRs (Totemeier, 2021).

Compared with oxide fuels, metallic fuels (U-Pu-Zr) have very high thermal conductivity, but they can experience high swelling and melt at a relatively low temperature (1,160°C). A typical fuel design is shown in Figure 4.2, where the metallic fuel slug is contained within cladding, with the two components being bonded by liquid sodium to improve heat transfer while accommodating the differences in thermal expansion between the metallic fuel and cladding (FRWG, 2018). However, given the risk of cladding failure, metallic fuels are incompatible with lead coolant because of their solubility in lead. Fabrication of metallic fuels, as outlined in Figure 4.3, involves (1) preparing the fuel feedstock from ore or spent nuclear fuel; (2) reducing the feedstock to metallic form; (3) alloying and casting by arc melting, vacuum induction melting, or microwave melting; (4) thermomechanical processing via rolling and coextrusion to remove defects; and (5) heat treatment to obtain the mechanical and material properties necessary to achieve the desired fuel performance (LaHaye and Burkes, 2019; Wood et al., 2020). Metallic fuel fabrication processes can generate large quantities of unrecoverable scrap (e.g., from the reaction of molten fuel with crucibles and fuel molds during the casting process) (see Carmack et al., 2017), which decreases resource utilization and provides challenges for safeguarding the material (Moore and Severynse, 2020). The Environmental Impact Statement for the Versatile Test Reactor assumes that up to 27 percent of the metallic fuel feedstock could be lost as waste (DOE-NE, 2022b). All steps of the metallic fuel fabrication process introduce pyrophoricity, chemical, radiation, and criticality hazards, as detailed in LaHaye and Burkes (2019).

A 2019 workshop identified near- and long-term research needs for the development and use of uranium-zirconium-based metallic fuels (Aitkaliyeva et al., 2020). The near-term research gaps related to swelling and fission gas release, fuel-cladding chemical interactions, phase diagram development and characterization, and thermal conductivity during reactor operation. The long-term needs identified to optimize fuel performance were better understanding of fuel creep/plasticity for individual fuel phases and lanthanide transport in the fuel during reactor

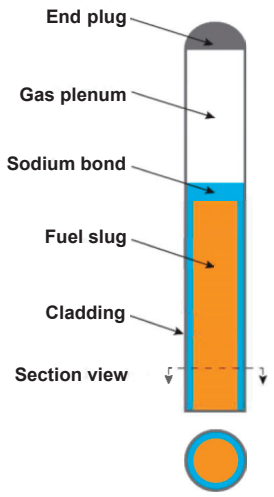


FIGURE 4.2 Schematic of a typical metallic fuel element sodium bonded to the cladding for improved heat conduction. SOURCE: Adapted from FRWG (2018). Courtesy of Oklo, Inc.

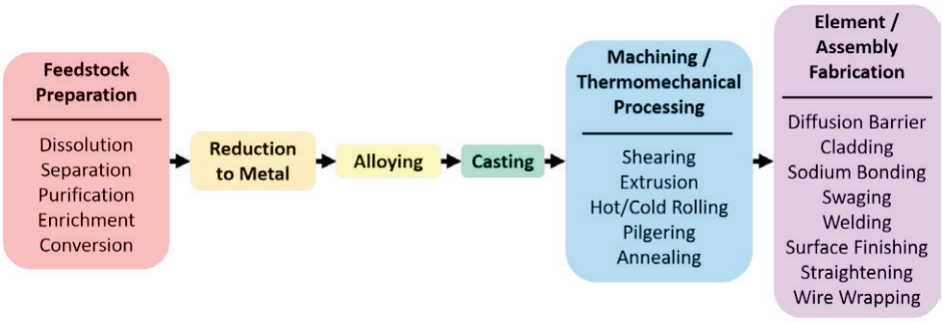


FIGURE 4.3 Process flow for fabrication of metallic fuel.
SOURCE: LaHaye and Burkes (2019). Courtesy of Pacific Northwest National Laboratory.

operation. For each gap, the workshop participants recommended experimental and modeling studies to obtain the requisite data for addressing the research needs. To facilitate further study of metallic fuels, the participants also recommended combining all existing data on such fuels—from experience at EBR-II, the Fast Flux Test Facility, and the Transient Reactor Test Facility, especially—into a single database.

The Advanced Fuels Campaign (AFC), part of DOE's Nuclear Fuel Cycle and Supply Chain Program, is focusing its efforts related to advanced reactor fuel development on metallic fuels, given their proposed use in several advanced reactor designs (INL, 2021b). Its primary goals in this area include qualifying existing metallic fuel designs and developing and qualifying sodium-free metallic fuel. Another key focus is establishing a licensing basis for metallic fuel, which will require compilation of existing data, as well as additional testing and data collection. The AFC aims "to develop and establish the qualification basis for a sodium-free metallic fuel design with extended temperature performance by 2027," noting that this initiative will benefit from collaborations with industry stakeholders and other DOE programs (INL, 2021b).

4.2.4.3 Other Advanced Reactor Fuel Types

Other fuel types being considered for advanced reactor designs include nitride fuels, mixed oxide (MOX) fuels, carbide fuels, and molten salt liquid fuels.[6] For a discussion of Th fuels and Th-based fuel cycles, see Section 3.2.5 in Chapter 3. Considerations for the fabrication and use of each of these fuel types are discussed below. In general, the production of both nitride and carbide fuels is more complex than that of MOX or metallic fuels. Additionally, fabrication of fast reactor fuels incorporating minor actinides (which could be present in reprocessed fuel materials; see Section 4.3.5 below) may require a much higher degree of complexity and introduce additional occupational safety considerations. These complications are illustrated in Table 4.1 later in this chapter.

Nitride fuels (UN-PuN) are being researched in the United States, Russia, Japan, and Sweden. Their high melting temperature (2,762°C), high thermal conductivity (10 times that of oxide fuels), and high density of fissile atoms enable the potential advantages of larger power uprates, longer fuel cycles, and higher burnup (Wood et al., 2020). Drawbacks include low oxidation resistance and poor hydrothermal corrosion resistance, but additives can help to increase corrosion resistance. The presence of ^{14}N, the most abundant natural isotope of nitrogen (99.6 percent), can negatively affect fuel performance and result in production of ^{14}C, whose half-life of 5,700 years can pose a radiation hazard. Using nearly pure ^{15}N can improve fuel performance and avoid ^{14}C production but requires expensive isotopic enrichment processes that are currently infeasible at scale (Wood et al., 2020). Russia's High Technology Research Institute of Inorganic Materials reportedly "has patented a technique for enrichment in ^{15}N, annual demand for which is expected to be several tonnes" (WNA, 2021d). Also, as mentioned in Section 3.2, LeadCold has stated that it has developed plans for acquiring enriched ^{15}N and manufacturing nitride fuel (Wallenius, 2021). Without prospects for large demand, these specialty enriched materials will likely remain expensive.

[6] See Table 3.1 for the fuel types being proposed by specific advanced reactor developers.

A number of methods for synthesizing nitride fuel have been studied, with the two most common being carbothermic reduction of uranium oxide followed by nitridation (CTR-N) and the direct thermal hydride-dehydride-nitride (H-D-N) synthesis (Wood et al., 2020). The CTR-N method is the only nitride fuel fabrication technique demonstrated at large scale; however, its requirements for large volumes of nitrogen and high temperatures, as well as long reaction times, lead to high processing costs and create challenges with working in inert atmosphere glove boxes. The H-D-N synthesis requires less nitrogen and shorter reaction times than CTR-N, but the resulting UN powder has a propensity for oxidation and therefore must be handled in moisture- and air-free environments, making process scale-up challenging. Following any synthesis method, the UN powder is consolidated to produce a dense fuel form. This process is more challenging with nitride fuels than with oxides or carbides, but several successful methods have been reported (Wood et al., 2020).

Mixed oxide fuel (UO_2-PuO_2) consists of a mixture of PuO_2 and depleted or reprocessed UO_2 powders that have been sintered into pellets, inserted into Zr-alloy fuel rods, and bundled into assemblies. It has low thermal conductivity and a low density of fissile atoms, but it does not react with Na or Pb. It is the reference fuel form for several reactors. For future advanced reactors, this fuel form could potentially be useful for and used in fast reactors, which would require a higher content of plutonium in their MOX fuel as compared with MOX fuel for LWRs. Chapter 2 provides more details about MOX for LWRs, and this subsection gives some brief relevant information about MOX fuel fabrication. Experiments have demonstrated that the oxide fuel is able to reach very high burnups: in the reactor Rapsodie, a burnup of above ~240 Gigawatt-days per metric ton of heavy metal was achieved in the 1970s. Commercial production of MOX fuel occurs at the Melox plant in France using PuO_2 from the La Hague reprocessing facility. This MOX fuel is distributed for use in 25–30 reactors worldwide, which operate on 30–50 percent MOX in combination with standard UO_2 fuel (Orano, 2021a; WNA, 2017a). Japan is currently constructing a MOX Fuel Fabrication Plant, scheduled to be completed in 2024, that will also use PuO_2 obtained by reprocessing spent UO_2 fuel (JNFL, 2020). As discussed in Chapter 2, U.S. efforts to construct a MOX fuel fabrication facility at Savannah River National Laboratory were unsuccessful, as the project faced significant cost, schedule, and budget challenges; it was terminated in FY2018 (Holt and Nikitin, 2017).

Carbide fuels (UC-PuC) have "high thermal conductivity and a high density of fissile atoms, but high swelling and poor compatibility with air and water" (Wood et al., 2020). The several types of carbide fuels include UC, UC_2, U_2C_3, and (U, Pu)C. These fuels are most commonly synthesized via carbothermic reduction (CTR) using a UO_2 precursor, but arc melting synthesis with uranium metal and graphite is also being explored (Wood et al., 2020). Research on UC fuels is being pursued in India, and India's fast breeder test reactor has run on mixed carbide fuel (70 percent PuC, 30 percent UC) since 1985 (Kumar et al., 2011).

Molten salt liquid fuels, in which the fissile or fertile material is dissolved in a molten chloride or fluoride salt, have been studied since the Molten Salt Reactor Experiment at Oak Ridge National Laboratory in the 1960s. The fuel fabrication syntheses will vary for the different reactor core designs proposed. As an example, Terrestrial Energy plans the following fuel fabrication process for its Integral Molten Salt Reactor: reduction of enriched UF_6 (4.95 percent ^{235}U) to UF_4 with hydrogen gas, followed by mixing with commercially available fluoride salts and further processing to remove impurities, oxides, and moisture (Terrestrial Energy, 2021). Many molten salt fuel syntheses will involve electrochemical or pyrochemical methods, as these are considered the most viable for industrial-scale processing (Wood et al., 2020). In all cases, molten salt fuel fabrication must be performed under an inert atmosphere, as the salts form corrosive acids in the presence of water.

4.3 BACK END OF THE FUEL CYCLE

The remainder of this chapter discusses fuel cycles that support, or conceptually could support, the existing LWR fleet and potential future advanced reactors starting with the reference case, the once-through fuel cycle. The introduction of fuel reprocessing in a few countries has resulted in the monorecycling of both uranium and plutonium in existing LWRs. However, fast reactors that would be able to use uranium and plutonium extracted from spent LWR fuel have not become available in the expected time frame, resulting in a need to mitigate the buildup of the extracted plutonium. The deployment of fast reactors in combination with advanced reprocessing

technologies would close the fuel cycle with the promise of efficient consumption of fertile uranium-238, multirecycle of plutonium, and concurrent management of the minor actinides. The goals of the multirecycle strategy are to (1) allow full utilization of the fertile uranium-238 by transmutation into plutonium-239 fuel; (2) multirecycle plutonium-239 to regenerate plutonium-239 at the expense of uranium-238; (3) remove or partition elements of interest—most notably, the minor actinides from the waste stream during spent fuel reprocessing; and then (4) destroy or transmute these elements into shorter-lived or stable species by recycling them in transmutation devices, such as advanced fast reactors.

While the goal of multirecycling plutonium is to maximize the energy extracted from nuclear fuel and subsequently natural uranium, the goal of partitioning and transmutation (P&T) endeavors is to minimize the alpha-bearing, radiotoxic, and heat-generating transuranic elements in the wastes requiring geologic disposal. In a multirecycle, fully closed fuel cycle scenario, if advanced reactors and fuels cycles supporting them could separate (partition) all of the actinides from spent fuel (uranium and transuranic elements) via reprocessing and transmute them (burn or fission them in advanced reactors) such that they are completely removed from the waste stream going to a geologic repository (no process losses), then all that is left is radioactive fission products (both short-lived, such as ^{137}Cs and ^{90}Sr/^{90}Y, and long-lived [see the table in Sidebar 1.2 in Chapter 1]). These radioactive fission products would require management and isolation from the biosphere (geosphere) for long periods of time. In the limit of P&T with 100 percent efficiency (a highly unlikely situation because processes are not perfect), the remaining high-level waste would be just radioactive fission products. As the efficiency of partitioning and transmutation decreases, due to process losses during partitioning and incomplete reactions during transmutation, more actinides become part of the high-level waste, which at late times adds to the heat load and the radiotoxic inventory of the high-level waste.

4.3.1 Introduction to Advanced Fuel Cycles

Advanced fuel cycles seek to maximize the utilization of natural uranium resources and reduce the high-level waste burden on permanent geologic repositories while maintaining the economic viability of power generated by nuclear systems and conforming to the highest levels of safety and nonproliferation. This is a tall order, and in practice, these goals will be extremely challenging to achieve simultaneously.

As noted in Chapter 1, Nuclear Energy Agency of the Organisation for Economic Co-operation and Development (NEA-OECD) describes advanced nuclear fuel cycle options in terms of a simple basis set of three fuel cycles: (1) once-through, (2) monorecycle, and (3) multirecycle options. Fuel cycle options can be further delineated as open, partially closed, and fully closed depending on the types of material sent for permanent disposal in lieu of recycling. In its 2010 report *Advanced Nuclear Fuel Cycles—Main Challenges and Strategic Choices*, the Electric Power Research Institute (EPRI) defined open, partially closed, and fully closed fuel cycles with respect to the management of Pu, noting that these terms are "often associated with different understandings by different authors" (EPRI, 2010b). EPRI's definitions follow:

- An "*open fuel cycle* is a fuel cycle in which Pu [spent fuel] is sent to a geologic repository for permanent disposal"; they include the once-through and the mono-recycle fuel cycles, when the spent LWR-MOX fuel is eventually disposed of in a geologic repository.
- A *partially closed cycle* is a fuel cycle in which Pu is continuously recycled while the minor actinides are sent to a geologic repository for permanent disposal;[7] and
- "A *fully closed cycle* is a fuel cycle in which no Pu or minor actinides are sent to a geologic repository for permanent disposal," but are being continuously recycled (EPRI, 2010b).

[7] This applies for the situation where spent MOX is stored pending reprocessing for future use in fast-spectrum reactors. But if spent MOX is eventually sent to a repository, monorecycle of Pu is an open cycle. Partially closed means multirecycle of Pu, or closed with respect to Pu management.

Other entities define fuel cycles in other ways, such as the French program, which emphasizes the fate of uranium and plutonium.[8]

Because heat generation due to plutonium isotopes and their decay products can impact a geologic repository, the EPRI definition focusing on the disposition of plutonium has been adopted for the purpose of this report. Importantly, material losses will inevitably occur with reprocessing and in other parts of the fuel cycle; therefore, no cycle can be perfectly closed.

The simplest and most straightforward fuel cycle is the once-through cycle that uses low-enriched uranium in LWRs and directly disposes of the spent nuclear fuel in a deep geologic repository. The once-through cycle, shown in Figure 4.4, is an open fuel cycle and is the reference to which all other fuel cycles are compared. As noted previously, the United States and most other nuclear power–producing countries—including Canada, Sweden, Finland, Switzerland, Spain, and Germany—have opted for the once-through cycle for their existing LWR fleets because of its relative simplicity, lower implementation cost in the short term, and anticipated higher proliferation resistance. No advanced facilities are required to support the once-through fuel cycle, as fuel reprocessing is not required. However, as indicated by Table 3.1 in Chapter 3, almost all advanced reactor developers intend to use a once-through cycle for the first decade or more of reactor operations.

4.3.2 Background on Reprocessing and the PUREX Process

Reprocessing refers to a set of operations or steps, both mechanical and chemical, that are required to remove the irradiated fuel from the fuel assemblies and cladding and to separate the spent fuel into process streams of reusable fuel material and waste. A reprocessing system or plant consists of four main components: (1) head-end processes (e.g., processes to separate fuel matrix from assembly structural materials and make the fuel form compatible with subsequent chemical separation steps); (2) chemical separation and conversion steps (e.g., processes designed to remove elements to be recycled and then convert them into a form compatible with the subsequent fuel fabrication process, such as oxide powder for aqueous-based processing); (3) waste management (e.g., off-gas treatment, fission product vitrification, metallic waste compaction); and (4) additional plant support systems (e.g., reagent and solvent recycle systems, methods for process control and accountability, robotics, in-cell maintenance capabilities).

Separations processes are the core of a reprocessing plant, as they are the chemical processes designed to specifically remove or isolate elements of interests based on their chemical properties from the mixture of elements in spent nuclear fuel. The separations are often difficult and complex because the mixture of elements in spent nuclear fuel represents roughly one-third of all of the elements in the periodic table (Nash and Lumetta, 2011; Taylor, 2015). The separation processes can be characterized broadly as aqueous- (hydro-) or nonaqueous-based (pyrochemical and pyroelectrochemical) processes.

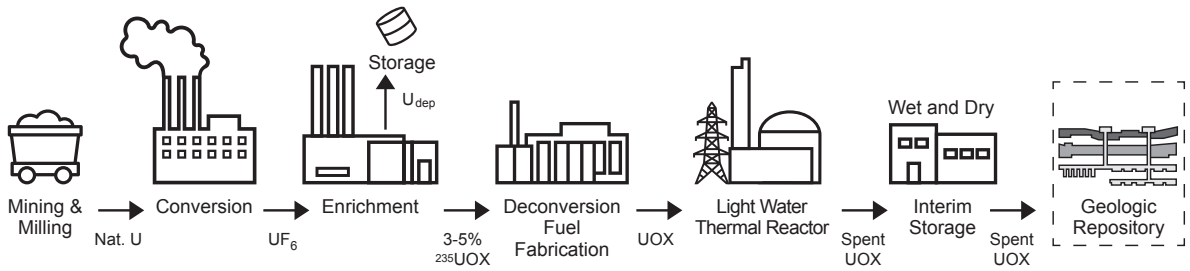


FIGURE 4.4 Schematic of an open or once-through fuel cycle for a light water reactor (LWR).
NOTE: Nat. U = natural uranium; U_{dep} = depleted uranium; UF_6 = uranium hexafluoride; UOX = uranium oxide.
SOURCE: Adapted from MIT (2011).

[8] In France, a *partially closed fuel cycle* is where spent nuclear fuels are reprocessed for recycling fissile and fertile materials (uranium and plutonium) to increase the energy production from the same initial material. Monorecycling where irradiated mixed oxide fuels are irradiated once and disposed of falls in this category. The French definition of a *closed fuel cycle* is where uranium and plutonium are continuously recycled, while the minor actinides can either be recycled or sent to a geologic repository for permanent disposal (Poinssot and Boullis, 2012).

To date, the only commercial-scale, industrially deployed process for reprocessing spent nuclear fuel is the well-known PUREX (plutonium uranium reduction extraction) process, a mature aqueous-based separation process[9] with an operational experience base of more than 60 years. PUREX has been the industry standard for separating uranium and plutonium from both oxide and metallic fuels. It involves the extraction, reduction, and back extraction of uranium and plutonium between a nitric acid solution and an organic phase containing tributyl phosphate. PUREX results in two separate pure products streams, uranium and plutonium, and a nitric acid solution containing the minor actinides (Np, Am, and Cm) and fission products. The nitric acid solution is designated as high-level waste and is further treated to immobilize the minor actinides and nonvolatile fission products in a glass matrix using vitrification. In the vitrification process, the radioactive materials are dispersed within the glass and chemically bonded in the glass network. While still hot, the vitrified high-level waste is poured into a standardized stainless-steel container and allowed to cool and solidify. Reprocessing also generates a solid waste stream mostly consisting of metallic fuel assembly hardware (e.g., fuel cladding and fuel assembly structural materials). At La Hague, these solid wastes are compacted to up to 65 percent of the metal density and placed in another standardized stainless-steel container. These uniform stainless-steel containers serve as the waste packages for both the high-level waste and metallic residues. The waste packages are designed to provide a safe, stable, and highly reliable compact form for long-term storage (>300 years) or until a geologic repository is available.[10]

Reprocessing spent fuel generates other process and operational wastes, including solid (residues, structural materials and equipment, resins, filters, and personal protective equipment), liquid (organic and aqueous), and gaseous (process off-gases) waste (CEA, 2008, 2009; NEA-OECD, 2006b). Each type of waste needs to be treated and conditioned in accordance with industry standards and applicable regulations for transportation, storage, or disposal for that particular waste category. These wastes will be low-level waste, as well as GTCC (Greater-than-Class-C waste). (See Appendix D for definitions of the various waste categories.) By volume, reprocessing produces more low-level waste and GTCC than high-level waste (ANDRA, 2006). (See Chapter 2 for more details and industry data about the amounts of these wastes from PUREX reprocessing.)

Since its inception, the PUREX process used for LWR fuel has been continually optimized for maximum recovery (>99.9 percent) and purification (decontamination factors versus fission products $>10^5$) of the uranium and plutonium product streams. These improvements have also led to a significant reduction in the volume of waste requiring geologic disposal, reduction of in-process water consumption, and the recovery and reuse of many of the reagents and solvents used in the reprocessing. Other advances have been incorporated into plant operations to increase process efficiency (e.g., redesign of liquid–liquid extraction equipment, such as mixer-settlers, pulsed columns, and centrifugal extractors for reliability and ease of maintenance), as well as to decrease secondary waste (e.g., optimization of process flow sheets and procedures). These improvements have been carried out concurrently while increasing plant safety and implementing many additional safeguards outlined by the International Atomic Energy Agency (IAEA) (CEA, 2008; Poinssot, 2021).

Future reprocessing plants are expected to face additional, more stringent, and costly requirements to improve safety and further reinforce safeguards, security, and proliferation resistance while reducing environmental impact and waste production. The cost-effectiveness of reprocessing and recycling is a controversial issue with studies presenting various conclusions, likely because of the difficulty of estimating and accounting for the cost of reprocessing and geologic disposal in the very long-term, as well as comparing short- (reprocessing) with long-term (disposal) costs. For more detailed information on the state of the art in reprocessing at La Hague, see CEA (2008, 2009).

As described in the next sections, implementing the mono- and multirecycle fuel cycles requires reprocessing.

4.3.3 Monorecycling of U and Pu in LWRs

Monorecycling of Pu in LWRs using PUREX reprocessing is the only fuel cycle besides the once-through fuel cycle that has been commercially deployed to support LWR operations (CEA, 2015). The U and Pu product

[9] PUREX was developed at Oak Ridge National Laboratory. See Long (1967).

[10] The waste packages for both high-level waste and metallic residues described are those used by the French Reprocessing program at La Hague, France, in CEA (2008).

streams from PUREX can be recycled as reprocessed uranium oxide ($U_{rep}OX$) fuel and MOX fuel, respectively.[11] As mentioned in Chapter 2, U_{rep} must first be reenriched before it can be recycled. Reenrichment of U_{rep} is performed in a dedicated cascade because of potential cross contamination by ^{232}U. An overenrichment of ~0.5 percent is required to counteract the presence of ^{236}U after enrichment.[12] Another option for reenrichment involves downblending by using some of the existing stockpiles of excess-defense-related highly enriched uranium. This approach is technically preferable because it limits the buildup of ^{236}U. Figure 4.5 illustrates the fuel cycle steps for monorecycling U_{rep} in LWRs, which provides ~8 percent savings in natural U resources compared with the once-through cycle[13] (EPRI, 2009b; IAEA, 2007a, 2009).

On the other hand, Pu can be mixed with U_{dep} and directly used to make MOX (U/Pu) fuel for recycle in the LWRs (LWR-MOX). Implementing a fuel cycle for the monorecycle of Pu requires the addition of a MOX fuel fabrication facility to the supporting infrastructure. The current industrial practice is to recycle LWR-MOX fuel only once in LWRs because of the buildup of nonfissile, even Pu isotopes. Figure 4.6 shows the steps for monorecycling Pu in LWRs.

Because LWR-MOX fuel typically contains 7–10 percent Pu, the fissile inventory and minor actinide content of spent LWR-MOX fuel is higher than that of spent UOX fuel.[14] As a result, spent LWR-MOX fuel exhibits characteristics additional to those of spent UOX that must be managed appropriately. These include a higher decay heat,[15] an increased potential for criticality, and (because of the increased minor actinide content) a higher

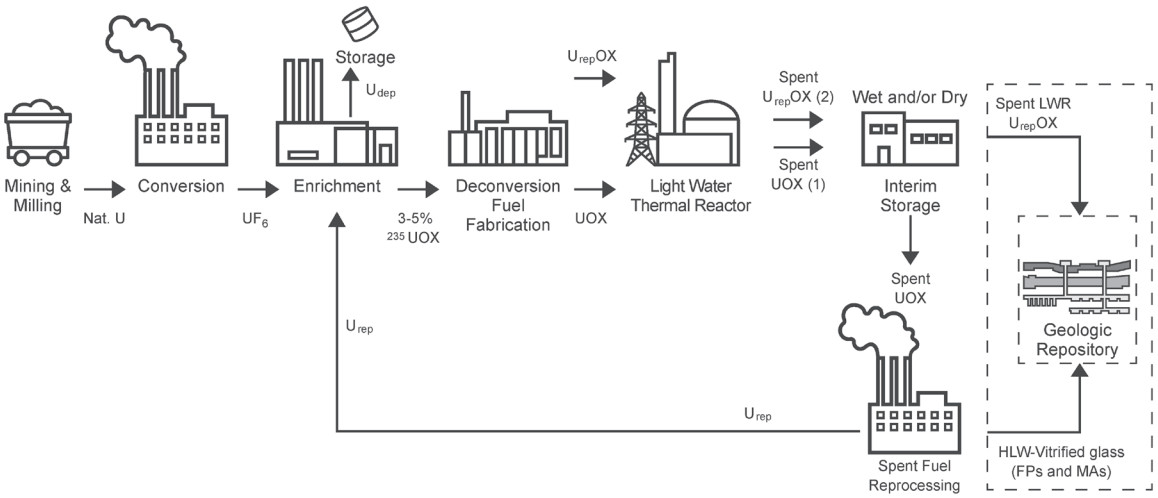


FIGURE 4.5 Steps in monorecycling uranium (U) in light water reactors (LWRs). Spent UOX (1) refers to the spent fuel resulting from the initial load of UOX fuel. Spent $U_{rep}OX$ (2) refers to the spent fuel resulting from the initial load of $U_{rep}OX$ fuel. NOTE: FP = fission product; HLW = high-level (radioactive) waste; MA = minor actinide; Nat. U = natural uranium; U_{dep} = depleted uranium; UF_6 = uranium hexafluoride; UOX = uranium oxide; U_{rep} = reprocessed uranium; $U_{rep}OX$ = reprocessed uranium oxide.
SOURCE: Adapted from MIT (2011).

[11] As implemented in France, reprocessing and recycling can be adjusted to mitigate any accumulation of separated materials. According to the French nuclear industry, the current accumulation of separated materials in the French program is temporary and related to the ongoing adjustment of the MOX manufacturing process to accommodate new U powder. See Table 2.1 in Chapter 2 and references therein about the accumulated stockpiles of separated Pu in France and other countries.

[12] This creates a constraint for plants that operate with fresh fuel enriched at >4.5 percent, because the higher reenrichment would bring the required enrichment value to >5 percent, which is over the enrichment limit of present LWR fuel fabrication plant licenses.

[13] Savings estimates can vary by a few percent depending on the assumed reference scenario (e.g., initial fuel enrichment, burnup, residual enrichment at discharge, assigned ^{236}U enrichment penalty).

[14] See Chapter 2, table in Box 2.1, where the masses of Pu isotopes and the main MAs in a total of *seven* spent LWR fuel rods are compared to one spent LWR-MOX rod 3 years after discharge.

[15] See Chapter 2, Figure 2.2, which compares decay heat of spent UOX to that of spent MOX fuel irradiated to the same burnup.

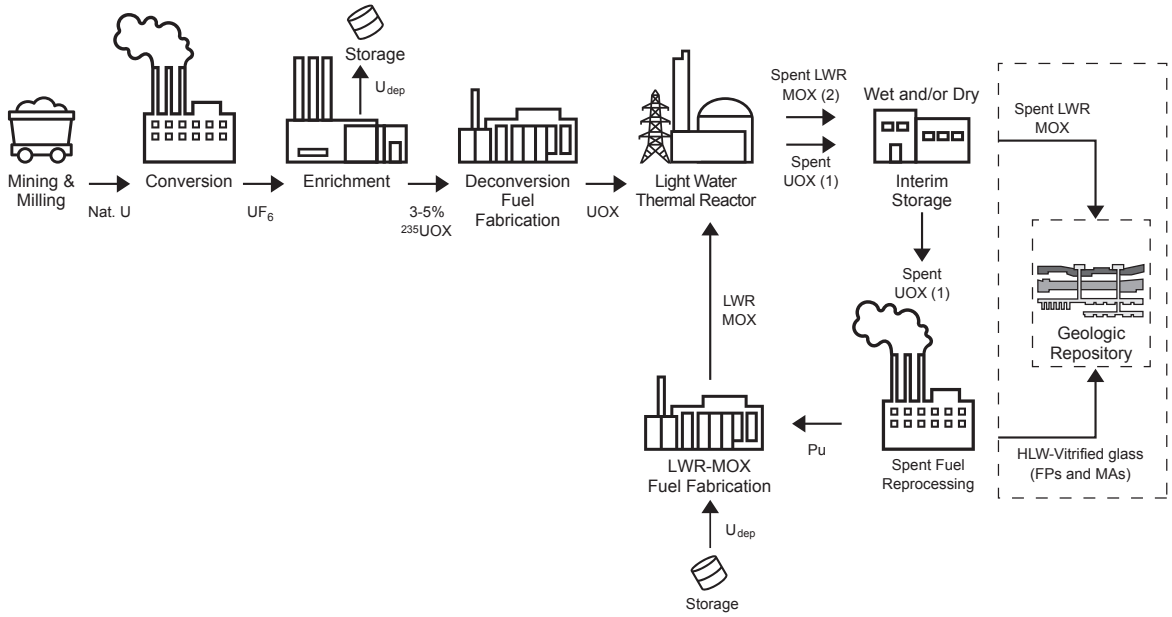


FIGURE 4.6 Steps for monorecycling Pu fuel in LWRs. Spent UOX (1) refers to the spent fuel resulting for the initial load of UOX fuel. Spent LWR MOX (2) refers to the spent fuel resulting from the initial load of LWR-MOX fuel.
NOTE: FP = fission product; HLW = high-level (radioactive) waste; MA = minor actinide; Nat. U = natural uranium; U_{dep} = depleted uranium; UF_6 = uranium hexafluoride; UOX = uranium oxide.
SOURCE: Adapted from MIT (2011).

radiation source term (mainly alpha, but gamma and neutrons as well). Because recycle of MOX into a fast reactor (FR-MOX) is not currently an industrially available option, spent LWR-MOX fuel is being stored, awaiting a final decision to either be recycled, should fast reactors become available, or to be disposed of in a geologic repository. Monorecycling of Pu in LWRs should be classified as "open," just like the once-through UOX fuel cycle, when the spent MOX fuel is not reprocessed and is instead disposed of as high-level waste in a geologic repository. When both U and Pu are recycled, this fuel cycle provides ~20 percent savings in natural resources compared with the once-through cycle (roughly 10 percent U and Pu each) (EPRI, 2010b; Poinssot and Boullis, 2012).

4.3.4 Fuel Cycle for the Multirecycling of Pu in LWRs

In the absence of fast reactors, a potential next step in the progression of fuel cycles could be multirecycling of Pu in LWRs. In this case, spent fuel (UOX and MOX) is always reprocessed, and the recovered Pu is, at least theoretically, indefinitely recycled in an LWR fleet, providing a path to stabilization[16] of spent fuel and Pu inventories.[17] However, multirecycling of only Pu in MOX-fueled LWRs has limitations, because the fissile Pu content decreases with each recycle. Compensating for the decrease in fissile Pu quality by increasing the overall pluto-

[16] The number of fuel assemblies decreases by a factor of 6–7 when spent UOX fuel is reprocessed. When MOX fuel assemblies are also reprocessed and the recovered Pu is recycled by adding fissile ^{235}U makeup, MOX is no longer stored cumulatively. At this point the number of spent fuel assemblies no longer increases, so all of the spent fuel is stabilized. How "fast" this situation will be realized depends on the implementation details (i.e., the rate at which reprocessing of the spent MOX and MOX-EU assemblies can be accomplished). Note that the description is relevant for space savings in the number of spent fuel assemblies that need to be stored either at a reprocessing facility or at a temporary spent fuel facility. The geologic repository footprint would have to take into account the additional heat load from the spent MOX fuel assemblies.

[17] La Hague has already reprocessed spent MOX fuels with a residual fissile content greater than that in low-enriched uranium fuel by using *dilution*, which consists of processing MOX (high residual fissile content) concurrently with LEU-UOX (low residual fissile content). With such an approach, the capacity for processing spent MOX fuel is only part of the total capacity of the plant. Reprocessing MOX fuels at industrial scale would therefore require adding workshops to the current reprocessing plant to avoid this dilution step.

nium content is limited to a maximum of ~12 percent Pu, at which value the void coefficient becomes positive (Martin et al., 2018). To overcome this limitation, multirecycling of Pu requires the addition of enriched rather than depleted uranium in the fabrication of the MOX fuel (LWR-MOX-EU). The realized savings in natural U do not differ significantly from what monorecycling can achieve, but multirecycling Pu in an LWR can provide a way to manage the inventory of Pu; it allows the extraction of additional energy while helping to minimize the buildup of Pu requiring storage.

While, in theory, no Pu would be sent for permanent disposal, there are always some losses during reprocessing. Compared with Pu monorecycling, there is an expected increase of ~70 percent in the accumulation of minor actinides going into the repository (Martin et al., 2018). Multirecycling of Pu constitutes a partially closed fuel cycle as classified by EPRI (2010b), in that the fuel cycle would be closed with respect to Pu (except for process losses) but open with respect to the minor actinides. As discussed in Section 2.3.2.2, U_{rep} is not currently being recycled in LWRs but rather stored, pending decisions on whether to recycle or dispose of it. Such a fuel cycle is shown in Figure 4.7.

France has recently announced its intention to assess the feasibility of such a fuel cycle (only for one or two additional recycling loops) as a means of stabilizing its spent fuel and Pu inventories as a transient stage before a potential deployment of fast neutron reactors (Landais, 2021). Similarly, Russia is working toward multirecycling Pu in LWRs via its REMIX program (described in Chapter 2).

4.3.5 Management of Minor Actinides

As early as the 1970s, studies, particularly in Europe, focused on the feasibility of a fully closed fuel cycle with multirecycling of plutonium and P&T of the minor actinides for the management of nuclear waste (Serco, 2011). Most fully closed fuel cycles using P&T have focused on a goal of a 100-fold reduction in the inventory of transuranic radionuclides in the resulting high-level waste, which implies recovery efficiencies greater than 99.9 percent (less than 0.1 percent process loss).

As first envisioned, P&T involved partitioning the minor actinides (and possibly select long-lived fission products) and placing them in some sort of transmutation device in which large quantities of neutrons react with these isotopes via fission or capture reactions to form shorter-lived or even stable isotopes. Four basic types of transmutation devices have been considered and evaluated: fast reactors, thermal reactors, accelerator-driven systems (ADSs), and fusion/fission hybrids (FFHs) (National Research Council, 1996). ADSs are hybrid accelerator/reactor devices in which a beam of particles from an accelerator strikes a subcritical target producing a burst of high-energy neutrons in a process called *spallation*. The neutrons interact with the isotopes to be transmuted in a subcritical blanket assembly causing them to fission. An FFH is also a subcritical reactor, but it differs from an ADS in that a fusion core provides the high-energy neutrons to fission the isotopes that are to be transmuted in the surrounding blanket. Consideration of the ADS and FFH technologies are not within the scope of this study, and they are not discussed further in this report. For additional information on ADS and FFH, see NEA-OECD (2002), IAEA (2015a), and DOE (2009b).

Neutron-induced fission is a transmutation reaction that, for minor actinides, results in much lighter, shorter-lived isotopes that would likely decay by beta emission or be stable. As discussed in Chapter 2, the neutron capture reaction in LWRs leads to the buildup of significant inventories of even-mass-number higher actinides (e.g., curium isotopes and even californium-252), because of their propensity for neutron capture (Carbonnier, 2006). Many of these isotopes, especially californium-252, are intense neutron emitters that would make reprocessing or fuel fabrication much more difficult—"the neutron source term for multi-pass recycling of MAs in LWRs is more than 2,000 times higher than that for fast reactors" (EPRI, 2010b). Consequently, transmutation of the minor actinides in thermal reactors would not be a very efficient process. This is not the case for fast neutrons, however, as shown in Figure 4.8.

With fast neutrons, all plutonium isotopes can fission and contribute to neutron production, although plutonium-239 and -241 provide the dominant contribution. The fission of even plutonium isotopes in a fast reactor has the added benefit of decreasing the production of minor actinides during reactor operation, whereas in thermal reactors, isotopes of minor actinides with even numbers of protons[18] continue to accumulate. Consequently,

[18] The sentence was corrected following release of a prepublication version of the report to clarify which isotopes of minor actinides accumulate in thermal reactors.

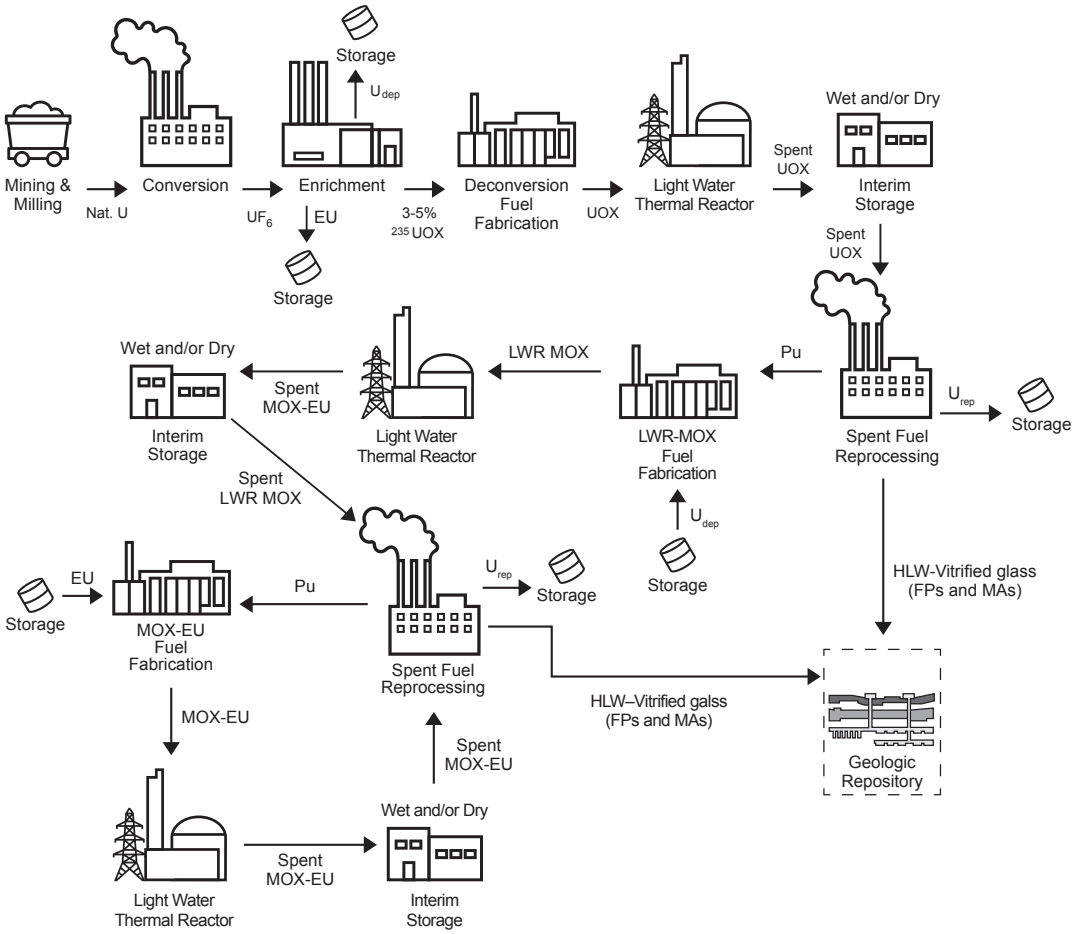


FIGURE 4.7 Multirecycling of Pu in light water reactors (LWRs) using mixed oxide fuel plus enriched U (MOX-EU).
NOTE: EU = enriched uranium; FP = fission product; HLW = high-level (radioactive) waste; MA = minor actinide; MOX = mixed oxide; Nat. U = natural uranium; U_{dep} = depleted uranium; UF_6 = uranium hexafluoride; UOX = uranium oxide; U_{rep} = reprocessed uranium.
SOURCE: Adapted from MIT (2011).

management of minor actinides has been evaluated primarily in the framework of closed fuel cycles with fast reactors (EPRI, 2010b). The P&T strategy for closing the fuel cycle by managing the MAs is a two-step process: (1) partitioning or isolating these elements in high yield and (2) efficiently transmuting them in fast reactors.

Closing the fuel cycle to recycle plutonium and destroy the minor actinides requires the development of advanced separation processes and fuel fabrication technologies. Fully closing the fuel cycle would require processing of advanced fuels that are much more radioactive than conventional uranium oxide fuel. For example, advanced fuels at discharge can have on the order of 100 times more decay heat and 1,000 times stronger radiation fields. These conditions are well beyond the current industrial experience base of reprocessing and fuel fabrication technologies, making it that much more difficult to perform these operations with low actinide losses, especially using aqueous reprocessing methods.

However, countries such as Russia and France have already gained limited experience on recycling fast reactor spent fuels. For example, the French program has demonstrated the capacity of recycling irradiated sodium-cooled fast reactor mixed oxide fuels by recycling more than 27 MT through their APM (Marcoule) and UP2-400 (La Hague) plants in the 1980s and 1990s (Poinssot and Boullis, 2012). In addition, France has acquired some experience with reprocessing, fabrication, and burning of spent fuel from the Phénix fast reactor

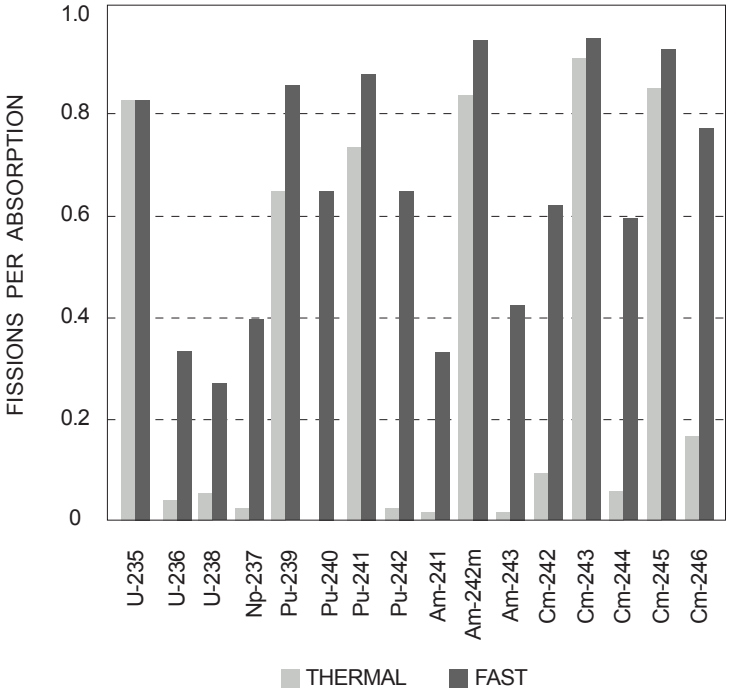


FIGURE 4.8 Probability of fission per neutron absorbed in actinide isotopes for thermal and fast spectra (e.g., for ^{239}Pu, probability 0.65 in a thermal spectrum and 0.87 in a fast spectrum).
SOURCE: Adapted from Wade and Hill (1997).

(Guidez, 2013, 2017). To achieve a 100-fold reduction in long-term hazards associated with spent nuclear fuel, fuel burnup in advanced reactors would need to be high, and reprocessing and fuel fabrication processes would have to be demonstrated on an industrial scale to have high recovery efficiencies with minimal (<0.1 percent) process loss. Dedicated R&D programs have been developed in Europe and Japan particularly to demonstrate the possibility of reaching such high recovery and recycling performances based on innovative and more efficient extractive molecules, such as monoamides (Mahanty et al., 2019; Prabhu et al., 1997; Taylor, 2015; Wang and Zhuang, 2019).

Also considered in early studies of minor actinide management was the possibility of P&T of some long-lived fission products (LLFPs), which are high-yield fission products with particularly troublesome behavior in the environment. LLFPs (including iodine-129, technetium-99, cesium-135, chlorine-36, and selenium-79) constitute only about 0.4 percent of spent fuel mass (compared with about 1 percent for plutonium and the minor actinides), but they are of concern for geologic disposal because of their predicted high mobility in the geosphere and biosphere due to their high water solubility, high diffusivity, and low sorption propensity. However, management of LLFPs is far less amenable to P&T techniques compared with that of minor actinides.

Because fission products do not undergo fission by thermal or fast neutrons, neutron capture reactions offer the only path for transmutation in reactors. Fission products are a direct consequence of the fission process and are relatively insensitive to whether thermal or fast neutrons initiate the fission reaction. Recycling is not a reasonable option for fission products, as they do not produce energy or add to the neutron inventory in the reactor. In fact, the result is just the opposite—transmutation of fission products is a net consumer of neutrons and thus a drain on the neutron economy in any type of reactor, fast or thermal. In theory, a transmutation strategy applied to LLFPs could be cost effective if neutron capture reactions (followed by beta decay) would lead to shorter-lived or stable transmutation products without having to first perform the costly step of isotope separation (Kailas et al., 2015).

In practice, efforts to transmute the LLFPs with either thermal or fast neutrons have been difficult due to a combination of their relatively small neutron capture cross sections and the associated isotopic compositions for these LLFPs.[19] However, research carried out by the French Atomic Energy Commission (CEA) shows that LLFP extraction is technically feasible by modifying the PUREX process; this is subject to important limitations and considerations (Zaetta et al., 2005).

With neutrons, only the transmutations of iodine-129 and technetium-99 have been considered seriously because of their large isotopic abundances and relatively large thermal capture cross sections (Chiba et al., 2017; EPRI, 2010b). CEA studies indicate that if iodine could be transmuted, its chemical form would be unstable under irradiation in a reactor. Thus, P&T of iodine-129 appears to be an unattractive option for the current technology. France, which supported a large national research program on the transmutation of LLFP from 1991 to 2006, concluded the transmutation of LLFPs was not viable and closed the door to this alternative (Poinssot, 2021). Although partitioning of LLFPs might become feasible, it will most likely be restricted to the most challenging fission products (EPRI, 2010b).

4.3.5.1 Plutonium Burning and P&T of Minor Actinides in Fast Reactors

As was previously mentioned, all plutonium isotopes (odd and even) in a fast neutron spectrum can fission and contribute to neutron production, although the plutonium-239 and -241 isotopes provide the dominant contribution. Furthermore, in fast reactors, the production of higher actinides, many of which are intense neutron emitters, is relatively minor compared with the buildup and accumulation of higher actinides during multirecycling in thermal LWRs. Consequently, any attempt to manage the minor actinides—that is, to reduce their mass for disposal in a geologic repository—only makes sense in the framework of closed fuel cycles with fast reactors (Szieberth et al., 2013; Tuček et al., 2008).

With the introduction of fast reactors, one possible scenario would involve the transition from an existing fleet of LWRs to a fleet of all fast reactors in a two-tier approach, coupling an LWR with a fast reactor for reprocessing the spent LWR-MOX fuel. The resulting plutonium would be recycled as fuel, and the minor actinides would be partitioned and transmuted in the fast reactor. As mentioned in Chapter 3, fast reactors can operate in a number of different modes based on the conversion ratio (CR), or ratio of fissile material produced versus that consumed. For this fuel cycle, the reactor is assumed to be operating as a burner reactor with a CR <1, thus it is consuming more fissile material than it produces. A schematic diagram of such a two-tier coupled LWR/fast reactor fuel cycle is shown in Figure 4.9, with the components of the fuel cycle before LWR-MOX fuel fabrication omitted for clarity.[20]

In 2009, a detailed time-dependent analysis of a two-tier coupled LWR/fast reactor fuel cycle (as shown in Figure 4.9) was conducted jointly by EPRI in the United States and Électricité de France (EDF), the French electric utility company. The objective of this study was to quantify the efficiency of FRs operating as burner reactors for recycling Pu and destroying the MAs for a coupled LWR/FR system (Machiels et al., 2009). The results indicated that, in addition to the decades needed to deploy advanced fast reactors and their associated fuel cycle facilities, many decades of continuous operation of the multirecycle fuel cycle are required to reach equilibrium for significant waste management benefits to be achieved. Furthermore, the results suggested that a more efficient approach than the two-tiered system would be to deploy fast breeder reactors that can produce new fissile fuel, thereby saving natural uranium resources as well as destroying the minor actinides.

Several countries, most notably France, Russia, China, and Japan, are developing transitional plans toward fully closed fuel cycles that first couple thermal LWRs with fast breeder reactors and advanced fuel cycles as a

[19] An LLFP is a specific isotope of a given element. Spent nuclear fuel most likely contains one or more isotopes of that element in addition to the specific LLFP. The isotopic composition is important because efforts to reduce the half-life of an LLFP should not lead to the creation of the same isotope or some others having longer half-lives through nuclear reactions induced on the other stable or short-lived isotopes of the element being transmuted.

[20] For simplicity, the components of the fuel cycle before LWR-MOX fuel fabrication in this figure have been omitted. The omitted parts are exactly the same as that shown in Figure 4.6 for monorecycling plutonium in LWRs with the exception that the spent LWR-MOX is no longer sent to a geologic repository but rather reprocessed to recycle plutonium and separate the minor actinides (neptunium, americium, and curium) for transmutation in a fast reactor.

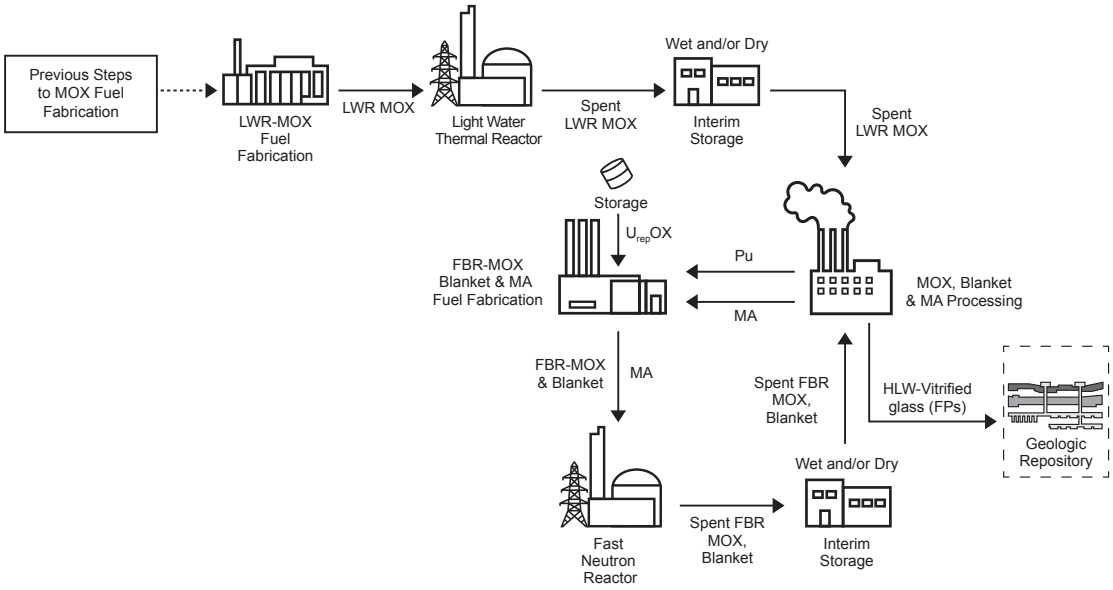


FIGURE 4.9 Schematic diagram of the fuel cycle that combines LWRs and fast reactors for multirecycling Pu and partitioning and transmuting minor actinides. If the fast reactor in the diagram is operating as a burner (CR <1), there would be no need for blanket fuel. Blanket fuel is required for breeding (where more fissile material is produced than consumed).
NOTE: FBR = fast breeder reactor; FP = fission product; HLW = high-level (radioactive) waste; MA = minor actinide; MOX = mixed oxide; $U_{rep}OX$ = reprocessed uranium oxide.
SOURCE: Adapted from MIT (2011).

means of recycling reprocessed nuclear materials. These transitional plans cover a time period of roughly 50–100 years over which thermal LWRs continue to generate electricity but are slowly phased out and replaced by advanced, new fast spectrum reactors (see Chapter 2, Section 2.5.1).

4.3.5.2 Plutonium Breeding and P&T of Minor Actinides in Fast Reactors

Breeder reactors operate with a CR ≥1 such that they both breed (via blanket fuel[21]) and burn plutonium and transmute (via burning) the minor actinides. Because breeder reactors both produce new fuel and can transmute minor actinides, they are the reactors of choice for P&T. In theory, a fleet of 100 percent fast breeder reactors would not need any additional uranium enrichment and would not consume any new natural uranium resources because they use existing depleted uranium stockpiles (depleted uranium tailings from previous enrichment operations) and would generate all of the plutonium needed for power production. The use of depleted uranium tailings would allow extending the current electricity production for several millennia (Poinssot and Boullis, 2012). The French program intends to pursue a strategy of operating fast reactors at a CR = 1 to develop a self-sustaining fast reactor fleet without any increase in the plutonium stockpile, while consuming the minor actinides.

For the case of plutonium breeding and P&T of the minor actinides in fast reactors, the same fuel cycle strategy as shown in Figure 4.9 is employed, except that the fast reactor is operating as a breeder/converter with a CR ≥1 and uses MOX fuel. The MOX fuel for the fast breeder/converter reactor (FBR-MOX) would have considerably higher plutonium content compared with LWR-MOX fuel (20–22 percent plutonium compared with 8–10 percent). The difficulty of reprocessing and recycling spent FBR-MOX increases with increasing plutonium

[21] Blanket fuel surrounds the reactor core in fast reactor designs and is composed of fertile material (e.g., uranium-238 or thorium-232) that forms fissile material (e.g., plutonium-239 or uranium-233) upon neutron capture. For the fissile material bred in the blankets to be used as new fuel, the blankets must be reprocessed to extract the fissile material from the remaining fertile material.

content, especially for aqueous processes, because of radiolysis and criticality concerns. With proper attention and procedures, these concerns may be manageable. Implementation of this fuel cycle requires advanced technologies (reactors, processing plants, and fuel fabrication facilities) that do not currently exist at industrial scale, although, as previously mentioned, reprocessing FBR-MOX has been demonstrated as feasible at significant scale of 27 MT in France (Poinssot and Boullis, 2012).

Depending on the fast reactor technology deployed, the fuel cycle for multirecycling plutonium and the P&T of minor actinides using an all–fast reactor fleet fuel cycle could look like that shown in Figure 4.10.

4.3.5.3 Homogeneous and Heterogeneous Recycling and Transmutation of the Minor Actinides

Two recycling approaches, homogeneous and heterogeneous, are being evaluated for the transmutation of minor actinides in fast reactors. These approaches differ in the location of the materials in the core and the manner in which they are irradiated (NEA-OECD, 2012).

Homogeneous recycling: In homogeneous recycling, the minor actinides, extracted by advanced reprocessing using either element-specific or grouped separations technology, are homogeneously diluted with uranium/plutonium as an integral component of the commercial nuclear driver fuel[22] (metallic, nitride, MOX, etc.). As an integral component of the driver fuel, the minor actinides can impact the reactivity and kinetic behavior of the core. Thus, loading minor actinides using this method will likely be constrained to no more than a few percent. This constraint also helps to limit the dose and decay heat that must be managed during fuel fabrication, transport, and handling. The experience base for homogeneous recycling is similar to that of MOX recycling, so reactor operations are not expected to be significantly altered by the homogeneous addition of a few weight percent of minor actinides to the driver fuel. Multirecycling of minor actinides in fast reactors generates higher actinides, although the accumulation is significantly less than that from LWRs. One major shortcoming is that this method contaminates the entire fast reactor fuel cycle with minor actinides. This disadvantage is mitigated by dilution, which avoids concentrating minor actinides in any one process or material form.

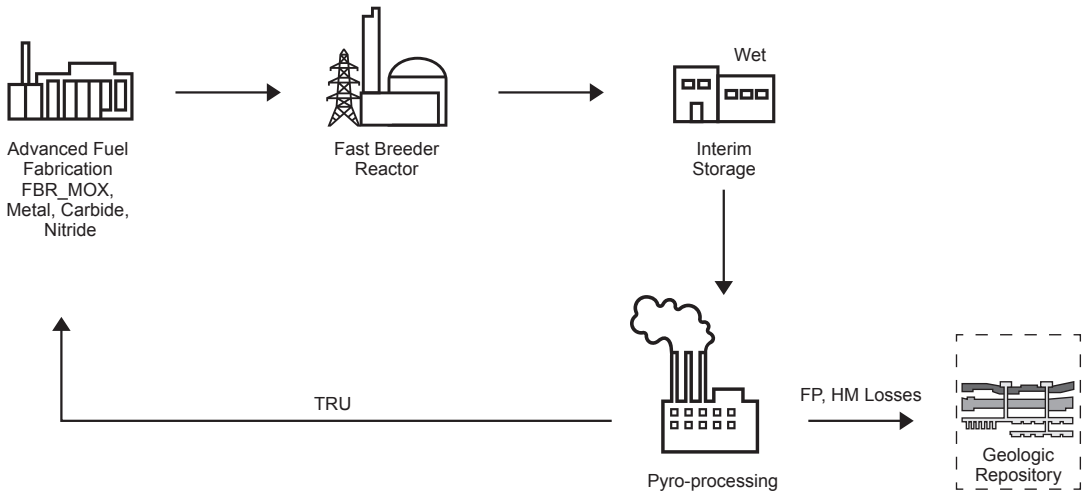


FIGURE 4.10 Notional fuel cycle for multirecycling plutonium and partitioning and transmuting minor actinides using an all–fast reactor fleet.
NOTE: FBR-MOX = mixed oxide fuels for fast breeder reactors; FP = fission product; HM = heavy metal.
SOURCE: Adapted from MIT (2011).

[22] Driver fuel, located within the reactor core, consists of the fissile material (e.g., used to sustain the nuclear chain reaction) and serves as the primary source of heat for electricity generation.

Transmutation of minor actinides in a homogeneous mode cannot be accomplished in a single pass but requires multirecycling. During the recycling, some higher actinides—many of which are intense neutron emitters—are generated, but this buildup is relatively minor compared with that from multirecycling in LWRs. The actual elemental and isotopic composition will be determined by which particular actinides are recycled. Notably, transmutation of minor actinides (Np, Np+Am, or Np+Am+Cm) does not eradicate them instantaneously. Their management and radiation exposure burden shifts to other parts of the fuel cycle, thereby making fuel fabrication, spent fuel reprocessing, and reactor operations more difficult. The extent of this challenge can be evaluated by determining the total inventory of transuranic elements by location at some arbitrary time in the fuel cycle. The inventory can be represented in terms of decay heat, γ dose, or neutron emission source term, as shown in Table 4.1 (Zaetta et al., 2005).

The values in Table 4.1 show factors by which a given problem (e.g., decay heat, γ dose, and neutron sources term) is magnified compared with a reference FR-MOX fuel during fuel fabrication and reprocessing. Content of neptunium, americium, and curium is increased by 2.5 percent compared with its content in the reference FR-MOX fuel. Based on the data in Table 4.1, neptunium does not present serious and unmanageable concerns for either fuel fabrication or reprocessing. Operations involving americium present greater difficulty during fuel fabrication because of the strong γ emission from neptunium-239 (daughter of americium-243), necessitating the use of shielded hot cells with remote handling systems to fabricate the fuel. Fuel fabrication is also difficult with curium recycling because of the high energy γ emissions from curium-243 and -244, and the neutron emissions from curium-244. In general, as seen in Table 4.1, the impacts of these minor actinides are much less significant for reprocessing than for fuel fabrication, because fission products dominate radiation exposure during reprocessing (EPRI, 2010b).

Heterogeneous recycling: Heterogeneous recycling involves loading minor actinides in special targets that are separate and distinct from the main driver fuel. This method requires selective element-specific separations (extractions) of minor actinides to place the preferred minor actinides in specific targets. To accomplish transmutation, these special targets containing relatively high concentrations of minor actinides (up to ~20 percent) are placed in the reactor to exploit excess neutrons produced by the main driver fuel. Due to the higher content of minor actinides, helium generation from alpha decay could be an issue affecting the behavior of the special targets during irradiation. The heterogeneous recycling method provides flexibility in that the location and residence times of the minor actinide targets in the core can be optimized for maximum transmutation.

Fabricating, reprocessing, and recycling of special minor actinide–containing targets are complex technical activities and would need additional development and demonstration to reach industrial-scale deployment. One strategy being considered for heterogeneous recycling that avoids the buildup of higher actinides is the complete burning of a target in a single cycle, rather than multirecycling, on a timescale similar to the fast reactor assemblies through the core. Such a deep burning strategy would require the use of advanced fuels and materials not yet available that are capable of withstanding high neutron doses (displacements per atom of >200) and harsh irradiation conditions for much longer periods of time than for typical fast reactor fuel assemblies. The downside

TABLE 4.1 Impact of Partioning and Transmutation of Minor Actinides on Fuel Fabrication and Reprocessing

Actinide Content of FR Fuel → (comparison with reference FR-MOX fuel)		2.5% Np	2.5% Am	2.5% Cm
Fuel Fabrication	Decay heat	× 1	× 4	× 12
	γ dose	× 4	× 80	× 500
	Neutron source	× 1	× 2	× 1,700
Reprocessing	Decay heat	× 2	× 3	× 6
	γ dose	× 1	× 1	× 1
	Neutron source	× 1	× 4	× 8

NOTE: FR = fast reactor; MOX = mixed oxide.
SOURCE: Adapted from Zaetta et al. (2005).

of this strategy is that if the deep burn is not successful (100 percent consumption of the target material), then the spent target fuel will become a waste problem (NEA-OECD, 2012).

Both homogeneous and heterogeneous recycling have advantages and disadvantages related to the reprocessing flow sheet, fuel fabrication and handling, and reactor operation and costs. Like all nuclear fuels that are to be recycled, fuels containing minor actinides must be robust with respect to the containment of fission products during and after irradiation and compatible with the capabilities of the reprocessing system used. Heterogeneous recycling tends to be more expensive than homogeneous recycling because it requires remote handling in more heavily shielded facilities for the fabrication of special target minor actinide fuel (NEA-OECD, 2018b).

4.3.5.4 Management of Minor Actinides: Conclusions

Implementation of advanced fuel cycles with plutonium recycling and the P&T of minor actinides requires the development of new actinide separation schemes that, in theory, may offer a wide variety of options for both material reuse and waste reduction. Plutonium and the minor actinides are produced in nuclear fuel under irradiation in either thermal or fast neutron spectra, and the buildup of each isotope is a strong function of burnup. As a result, spent nuclear fuel subjected to different burnup conditions can result in large isotopic variations (i.e., large differences in the ratios of isotopes of plutonium, neptunium, americium, and curium) that can have very different nuclear properties (Forsberg and Greenspan, 2003). Managing the minor actinides, in particular, is technically complex and depends on the specific objectives of the P&T strategy employed. The P&T objectives define which actinides are to be transmuted and lead to a large array of options, including the following:

- Partitioning:
  - Choice of reprocessing technology (e.g., aqueous- or nonaqueous-based, selective-element specific or a group separation of elements)
- Transmutation:
  - Method of recycling (homogeneous or heterogeneous, and if heterogeneous, single or multicycle)
  - Fuel fabrication technology (e.g., need for shielded hot cells equipped with automation and remote handling systems)
  - Fast reactor technology (e.g., sodium-cooled, gas-cooled, lead-cooled, or molten salt reactors) (e.g., the combination of reprocessing and the fast reactor technology chosen for transmutation)

As was concluded in the 1996 National Research Council report *Nuclear Wastes: Technologies for Separations and Transmutation* (National Research Council, 1996), the bottom line is that implementing a fully closed, multirecycle strategy of plutonium, and P&T of the minor actinides, requires a long-term and sustained commitment involving construction and operation of large, complex reprocessing facilities, equally complicated fuel fabrication facilities (because of higher levels of radioactivity), and transmutation devices such as advanced fast reactors. To realize the expected benefits of maximum use of uranium resources and a significant reduction in the waste source term due to the minor actinides requires the successful deployment and operation of not one, but all of these technologies (reactors, reprocessing, and fuel fabrication facilities). Aggressive pursuit of such a program is unrealistic to consider at this time, as it is inconsistent with current U.S. national security policies and does not have favorable economics. It would be of interest only in the context of a sustained national or international program with a time horizon of a century or longer.

4.3.6 Processing Advanced Reactor Fuels

Over the past several decades, research around the world has focused on the development of a vast array of novel separation processes to support various options for a closed fuel cycle. For aqueous processes (which are by far the most mature), ongoing R&D has focused on more efficient, simple (potentially single-cycle), and environmentally friendly processes that use salt-free aqueous and organic phases and employ highly selective ligands that contain only carbon, hydrogen, oxygen, and nitrogen (Baron et al., 2019). This R&D has further focused on

leveraging the current aqueous reprocessing experience by (1) allowing processing streams to be easily passed from one process step to the next with minimal process adjustments and (2) utilizing the same operationally tested liquid–liquid extraction contactor technology (mixer-settlers, pulsed columns, centrifugal extractors) in current use.

In addition to aqueous processes, nonaqueous (pyroprocessing) processes are also being investigated to handle fuels with higher plutonium content and shorter cooling times. The main countries participating in these efforts are the United States, Russia, the European Union, China, Japan, and the Republic of Korea (IAEA, 2008; NEA-OECD, 2018a). Several reviews published on the technical readiness levels (TRLs) of the various separation processes note that their technological maturity varies considerably (Baron et al., 2019; Collins et al., 2014; Joly and Boo, 2015; NEA-OECD, 2018a). Most separation processes are in the proof-of-principle range (TRLs 4–6), and still others are only at the proof-of-concept stage (TRLs 1–3). In addition, other processes are being developed that show promise for reducing waste by, for example, improving solvent and reagent recycling or recovering and recycling cladding material, such as zirconium. It is too early in technological development to determine which specific separations and recycle methods are achievable and could improve the economics and waste generation associated with fuel reprocessing.

As described previously, plutonium recovered from spent LWR oxide fuel can be recycled as mixed oxide fuel in LWRs (LWR-MOX). Reprocessing the spent LWR-MOX uses similar head-end and aqueous processes as spent uranium oxide fuel, with a couple of exceptions. During reprocessing of spent LWR-MOX, care must be taken to adjust the extraction process to account for a higher content of plutonium because of criticality concerns. Additionally, due to higher fuel burnup, the dissolution process may result in more intractable residues ending up as process losses that go to waste, although specific processes have already been developed in France to overcome this risk (Miguirditchian and Taylor, 2021; Miguirditchian et al., 2017). To date, ~70 MT of LWR-MOX fuel has been successfully reprocessed at La Hague (Todd, 2020).

Head-end processes would need to be developed for non–oxide-based fuels (e.g., nitride, carbide, metal, and metal-alloys) and TRISO fuel (see Appendix G). These advanced fuel types would have to be converted into standardized and convenient chemical forms compatible with subsequent aqueous or nonaqueous processing schemes, while minimizing process losses and efficiently capturing volatile and semivolatile off-gases containing fission products.

The concept of fuel reprocessing to support liquid-fueled and liquid-cooled molten salt reactors is somewhat different from that for solid-fueled reactors. Some molten salt reactors require an extensive chemical processing plant intimately coupled to the reactor. Head-end processes for liquid-fueled or liquid-cooled molten salt reactors differ in that they are used to prepare the spent nuclear fuel for subsequent use in molten salt reactors. Depending on the specific reactor type, fluorination or chlorination is used to convert spent oxide fuels to either fluoride or chloride salts. The head-end processes ensure the fuel salts are clean and free from impurities such as oxides, water, and potentially other oxidants (NEA-OECD, 2018b). The functions of chemical processing plants for reprocessing spent fuel salt for molten salt reactors are described in Section 4.3.6.5.

4.3.6.1 Advanced Reprocessing Strategies for Aqueous-Based Partitioning of the Actinides

Modifications to the aqueous reprocessing flow sheets for recycling of plutonium and partitioning the minor actinides are under development with the intent of preventing or significantly reducing the risks of nuclear proliferation. As a first step, these efforts have focused on separation schemes that do not produce a pure stream of plutonium that could be diverted for weapons use (DOE-NE, 2007). For example, the COEX process, developed by the French program, exploits evolutionary changes in the PUREX reprocessing flow sheet to address proliferation concerns. In this process, uranium and plutonium are processed together to produce a mixed solid solution of $(U,Pu)O_2$ and avoid having a pure plutonium product stream at any time during reprocessing (Paviet-Hartmann et al., 2011). Such coprocessing of uranium and plutonium also provides some advantages for the fabrication of mixed oxide fuel for recycle (Castelli et al., 2009). Variations of COEX are also possible where the coconversion can be done with either neptunium or americium ending up in the mixed oxide fuel—(U,Pu,Np)MOX or (U,Pu,Am)MOX—to be recycled (Drain et al., 2008). Solvent extraction processes, such as GANEX (Grouped ActiNide EXtraction), based on altogether different extractants are also being developed by the French program

with the goal of recovering all of the transuranic elements from a solution of dissolved spent fuel. Heterogeneous recycling, on the other hand, requires the development of even more advanced aqueous separation processes that add complexity to the reprocessing plant, as described below.

Assuming that neptunium can be managed as part of uranium/plutonium COEX or a similar process (e.g., UREX developed in the United States), the remaining waste stream contains americium, curium, and fission products. Americium, curium, and lanthanide fission products all exist as trivalent ions in solutions, making the chemical separation of americium and curium from the fission product waste stream challenging. Two-step processes have been developed to achieve such separations, such as DIAMEX-SANEX in France and TRUEX-TALSPEAK in the United States. In the first step, americium, curium, and the lanthanide fission products as a group are isolated from the other fission products; a second, more difficult, separation step recovers americium and curium as a group from the lanthanides. Individually separating americium from curium is yet an even more difficult task but potentially desirable, as removal of curium reduces the heat load and neutron source term during the fabrication of special americium transmutation targets. The French program has demonstrated the feasibility of the EXAm process for the isolation and heterogeneous recycling of americium (Poinssot et al., 2017a). The curium fraction is allowed to decay in storage, and after about 100 years, the decay products (mostly plutonium) can be either disposed of or recycled as mixed oxide fuel.

One of the most important technologies used in fuel reprocessing is the conditioning or processing of the waste prior to disposal. Currently, vitrification is the technology of choice for aqueous-based reprocessing of uranium oxide fuel, as it defines the resulting waste form in terms of its chemical composition (degradation behavior under near-field repository conditions) and its volume (limited by the heat-loading fraction in the specific glass composition). Conditioning technologies appropriate for advanced fuels will have to be developed and qualified if advanced reactors are deployed in the future, considering that borosilicate glass has already been demonstrated to be compatible with the high-level waste stream coming from the advanced reactors fuels reprocessing (CEA, 2009). Of particular note is the progress made over the past three decades by the French (CEA) R&D nuclear waste management program, which is studying potential benefits of aqueous fuel reprocessing to reduce both long-term volume and toxicity of wastes. The French program has established the "feasibility of homogeneously or heterogeneously recycling of the minor actinides in a Gen IV fast neutron reactor" using hot cell demonstration tests with actual solutions of spent fuel; however, to make these processes operational at commercial scale, more R&D is required (Poinssot et al., 2017b). Table 4.2 lists some of the more mature aqueous processes organized by element separated and the motivation for their development. Included in the table is the separation of cesium-137 (half-life ~30 years) and strontium-90 (half-life ~29 years), which along with their decay products, account for more than 90 percent of the decay heat from all of the fission products. If cesium and strontium were to be separated, they would likely be stored and managed separately from the repository.

4.3.6.2 Advanced Reprocessing Strategies for Nonaqueous, Pyroprocessing-Based Partitioning of the Actinides

Nonaqueous, or pyroelectrochemical, processes used for recycling spent nuclear fuel rely on refining techniques[23] conducted in molten chloride (or fluoride) salts at elevated temperatures (500–900°C). Countries currently engaged in pyroprocessing R&D programs include France, India, Japan, the Republic of Korea, Russia, and the United States (IAEA, 2021d). There are two primary pyroelectrochemical processing methods: (1) electrorefining, in which an electrical current of sufficient voltage separates elements in a molten salt by anodic dissolution concurrently with metal deposition at the cathode, and (2) liquid–liquid reductive extraction, or separations based on the selective reduction and extraction of a metal from a molten salt phase into an immiscible liquid metal phase containing the reductant (NEA-OECD, 2018b; Rodrigues et al., 2015). Both electrorefining and liquid-liquid reductive extraction can be used for either metallic or oxide fuels if an appropriate head-end treatment is added to the flow sheet.[24] Electrorefining of oxide fuel requires that the oxide fuels are first reduced to the metallic state, which

[23] Refining techniques are methods developed for reducing impurities in a substance, in this case a metal.

[24] This paragraph was modified after release of a prepublication version of the report to correct details concerning electrorefining and liquid-liquid reductive processes.

TABLE 4.2 Summary of Some of the More Mature Aqueous Separations Under Development to Support Various Fuel Cycle Options

Category of Separation	Element(s) Separated/ Recovered	Example Process(es)	Motivation
U-only	U	GANEX (1st cycle), NEXT UREX	Recover remaining fissile U for possible reenrichment
Major actinide corecovery	U and Pu separately	PUREX—Industry standard	Recover fissile U for reenrichment and Pu for recycle as MOX fuel
	U/Pu or Pu/Np as a group	COEX, UREX, UREX+	Aid MOX fuel fabrication and recycle
	U/Pu/Np as a group	TBP extraction in NEXT process	U/Pu/Np mixed oxide fuel recycle, send Am/Cm to waste with lanthanides and other fission products
Transuranic actinide separation	Pu/Np/Am/Cm	GANEX (2nd Cycle)	Group extraction of transuranic elements for homogeneous fast reactor transmutation
	Lanthanides/actinides as a group	DIAMEX, TRUEX	Remove trivalent lanthanides and actinides from PUREX raffinate
	Lanthanides/actinides separation leaving Am/Cm as a group	DIAMEX-SANEX, TRUEX-TALSPEAK	Avoid high neutron absorbers in fuel (La-Tb and Y) allowing heterogeneous recycling of Am/Cm as a group; lanthanides sent to waste
	Am from Cm	EXAm	Heterogeneous recycle of Am without the fuel fabrication problems related to decay heat and SF neutron issues with Cm
Cs/Sr[a] separation	Cs/Sr as a group	Cs Treat, Sr Treat	Separately manage short-term heat generators outside of a deep geologic repository; Cs, Sr Treat are decontamination processes in use for Cs/Sr-contaminated equipment

[a] Cs-137 and Sr-90 with their daughter decay products Ba-137 and Y-90 alone provide over 90 percent of the decay heat from all fission products.

SOURCES: Baron et al. (2019); NEA-OECD (2018b); Okada (1985); Rodríguez-Penalonga and Moratilla Soria (2017); Vandegrift et al. (2004).

can be done by chemical or electrochemical methods, before being subjected to the normal pyroelectrochemical processes used with metallic fuel.[25] To date, no commercial-scale pyroprocessing facilities have been built, but two chloride salt–based, pyroelectrochemical metallurgical processes have been implemented at pilot "engineering" scales: (1) electrowinning[26] of oxide fuels at RIAR in Russia and (2) electrorefining of metals, pioneered by Argonne National Laboratory and ongoing since 1996 for the electrometallurgical treatment of metallic spent EBR-II reactor fuel[27] (National Research Council, 2000; NEA-OECD, 2018b).

Other nonaqueous methods, such as those based on fluoride (and chloride) volatility, have also been investigated and are being considered for online fuel reprocessing in molten salt reactors (NEA-OECD, 2018b; Pereira, 2020). Also see Section 4.3.6.4 for a discussion of fluoride-based volatility methods for molten salt reactors.

[25] Prior to reduction of uranium oxide fuel, an additional process step called "electrowinning" (see next footnote) can be carried out to significantly reduce the volume of material to be treated, as is being developed in Japan (NEA-OECD, 2018b). In this step, uranium dioxide is electrolytically dissolved from an anode basket and reelectrolyzed as "pure" uranium dioxide on the cathode. The pure oxide is converted to metal, typically by an electrochemical reduction step, and then further processed using electrorefining.

[26] Electrowinning is a generic term for electrodeposition of a species from an electrolyte solution onto a cathode and is accomplished by applying a potential across two electrodes to separate the species based on its reduction potential.

[27] The initial kg-scale electrometallurgical treatment operations on spent EBR-II fuel were carried at Argonne-West, which later became part of the Idaho National Laboratory.

4.3.6.3 Electrometallurgical Treatment of Sodium-Bonded Spent Nuclear Fuel

Argonne's electrometallurgical treatment (EMT) used to treat EBR-II fuel[28] is described briefly to illustrate pyroprocessing of metallic uranium spent fuel.[29] Key to the EMT process is the electrorefining step. Broadly, *electrorefining* involves the anodic dissolution of metallic fuel into a molten salt electrolyte (typically, an LiCl-KCl eutectic mixture) followed by, depending on the electrode material used and the cell potential, the electrodeposition of uranium only onto the cathode. The driving force for the separations is the free energy of formation (ΔG_f°) of the various metal chlorides in the spent nuclear fuel. When a constant controlled current is allowed to flow between the anode basket containing chopped fuel elements (including the stainless-steel cladding) and the cathode, the uranium, plutonium, transuranic elements, alkalis, alkaline earth, and rare earth (lanthanide) metals oxidize and dissolve into the molten salt as cations. Left behind in the anode basket are elements such as the noble metals (small ΔG_f°, least stable chlorides) that do not dissolve.[30] The alkalis, alkaline earth, and rare earth (lanthanides) elements form very stable chlorides (large ΔG_f°) that are easily oxidized but not easily reduced, so they remain in the salt. The actinides (intermediate ΔG_f°) are efficiently transported in the molten salt to the cathode. By controlling the electrode material and redox potential, metals can be reduced and deposited on the cathode. In the case of the EMT process, the anode and cathode potential are controlled to allow only the reduction of U^{3+} to uranium metal at the cathode. Because salts tend to be hygroscopic and metals tend to be pyrophoric, pyroelectrochemical (electrorefining) processes must be carried out under an inert atmosphere in which the levels of oxygen and water are tightly controlled, adding another level of complexity to an industrial-scale pyroelectrochemical processing plant.

A schematic of the EMT processes used to recover relatively "pure" uranium is shown in Figure 4.11. This process uses a steel cathode to separate "pure" uranium. Further purification of the uranium occurs in a cathode processer and involves the removal of residual salts by distillation. The final step in the process is to cast the uranium metal product into an ingot for storage. A variant of the process can be used to recover the transuranic elements. After most of the uranium has been deposited, the steel cathode is removed and replaced with a liquid cadmium cathode (LCC) on which the transuranic elements and remaining uranium (and some residual lanthanide fission products) can be codeposited at the desired cathodic potential. The adhering salt and cadmium can be distilled, leaving the actinides to be further processed and remotely fabricated into fuel to be recycled.

Two main waste streams are generated from the EMT pyroprocess. The first consists of the stainless-steel cladding hulls, which contain any unoxidized material, including fission products, and the noble metals (i.e., undissolved solids) remaining in the anode basket. After removing the adhering salt from these materials, some Zr is added to form a lower-melting alloy. The resulting material is cast into a Fe/Zr metal waste form. Periodically, some electrorefiner salt-containing transuranics, some residual uranium, alkali and alkaline earth fission products, and rare earth (lanthanide) fission products are removed and mixed with zeolite. (Methods are available to recover the uranium and transuranic elements from the salt prior to its disposal).[31] Upon heating, the salt reacts with the zeolite to form sodalite (e.g., zeolite + sodium chloride forms sodalite). Finally, glass is added, and the mixture is heated and undergoes pressureless consolidation to form a glass-bonded sodalite ceramic waste form. Like aqueous reprocessing systems, pyroprocessing-based systems must manage fission product gases such as xenon and krypton, as well as fuel assembly hardware that becomes activated during irradiation. Typically, the atmosphere control system for a pyroprocessing facility would be used to collect the fission product gases released during decladding and other head-end processes; activated fuel assembly hardware would be separated and compacted or melted to reduce volume for disposal most likely as high-level waste (Frank et al., 2015; IAEA, 2008; National Research Council, 1996, 2000; Williamson, 2020). For fuels with zircaloy cladding, because of the large volumes involved, other pyroprocesses are being developed to potentially recover and reuse zirconium.

[28] EBR-II driver fuel was approximately 65 percent–enriched in uranium-235, metallic fuel alloyed with ~10 percent zirconium, depleted uranium blanket fuel, and stainless-steel cladding. Sodium was used to bond the fuel to the cladding for better heat conduction.

[29] This sentence was altered after release of a prepublication version of the report to remove incorrect information about pyroelectrochemical processes.

[30] Noble metals demonstrate outstanding resistance to oxidation even at high temperatures. Although the group is not defined strictly, it usually includes the metals of groups VIIb, VIII, and Ib of the second and third transition series of the periodic table: rhenium, ruthenium, rhodium, palladium, silver, osmium, iridium, platinum, and gold. On occasion, mercury and copper are considered noble metals.

[31] Electrowinning methods can be used to recover the uranium and transuranic elements remaining in the salt after electrorefining (Choi and Jeong, 2015).

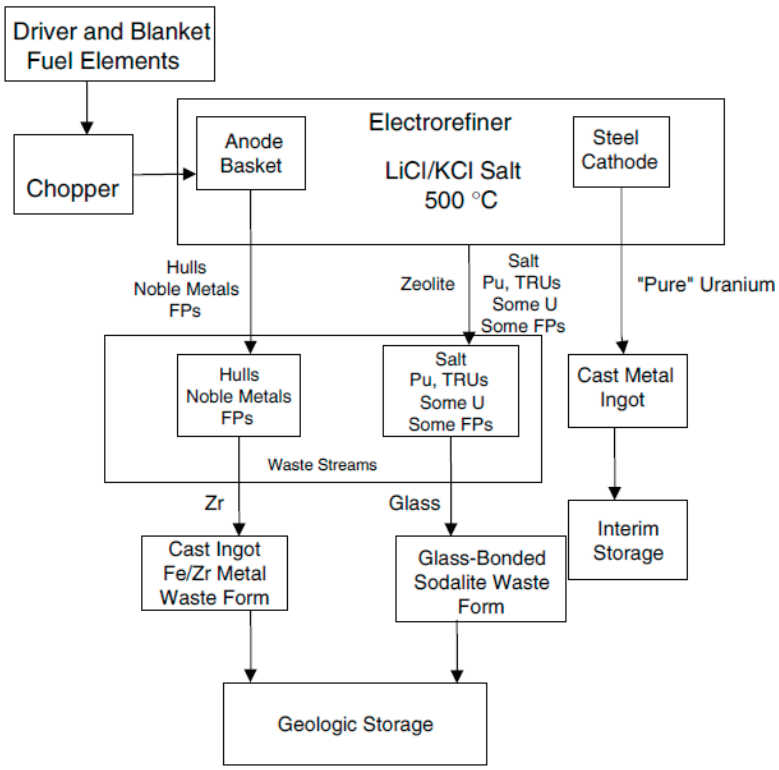


FIGURE 4.11 Schematic of pyroelectrometallurgical treatment used for metallic fuel for the Experimental Breeder Reactor II. Block diagram with (left) and without (right) the cadmium electrode. "Pure" uranium in the diagram denotes uranium that is relatively pure but requires additional purification steps to reach the desired purity.
NOTE: FP = fission product; TRU = transuranic element.
SOURCE: National Research Council (2000).

Efforts to pyroprocess sodium-bonded highly enriched uranium metallic fuel from both EBR-II and Fast Flux Test Facility (FFTF) using the EMT process have continued at INL in concert with Argonne National Laboratory since the end of the demonstration program in 2000 (Patterson, 2021; Williamson, 2020). Valuable lessons have been learned guiding further innovations throughout the EMT processes. Examples include the following:

- Process—elimination of alloy-forming liquid metal cadmium cathode in favor of a nonalloying solid metal cathode for (co)-recovery of uranium and transuranic elements; staggered-batch continuous-electrorefiner operations.[32]
- Processing hardware—scraped cathode to collect and consolidate recovered uranium more efficiently; new multifunctional furnace to reduce bottlenecks, improve efficiency, and provide redundancy in distillation and casting steps.
- Facility design—integration of facility, process, and safeguards, including increased automation.
- Safeguards—in situ process monitoring using voltammetry, input accountancy, and inventory modeling.
- Waste forms—promising iron-phosphate glasses with higher waste density.

Idaho National Laboratory also sees continued processing of EBR-II driver fuel as a near-term solution to the supply of HALEU for some of the advanced reactor concepts; see Section 4.2.3.1 for some of the challenges and limitations of this approach.

[32] This bullet point was corrected following prepublication release of the report to refer to (co)-recovery rather than recovery.

4.3.6.4 Other Nonaqueous Pyroprocesses

In addition to pyroelectrochemical methods (e.g., electrorefining and reductive extraction), nonaqueous processes based on fluoride or chloride volatility have been investigated (Uhlíř and Marecek, 2009). Halogenation with F_2 gas generates volatile hexavalent fluoride compounds of U, Pu, or Np that are then captured, condensed, and reduced to the tetrafluoride species for recycle.[33] Volatility methods are being considered in combination with electrochemical methods to process[34] salts in molten salt reactors (MSRs) (Pereira, 2020). Other processes under development include ion exchange, melt crystallization, vacuum distillation, oxidative precipitation, phosphorylation, and dehalogenation, to name a few (IAEA, 2021d). Many of these processes are also being evaluated for separating and recycling MSR-type salts (Riley et al., 2018, 2019). Most of these processes are in early stages of development and require considerably more R&D (Baron et al., 2019).

Another dry process for recycling spent fuel is the voloxidation process (or Atomics International Reduction Oxidation [AIROX]), which uses only gaseous and solid materials. In this process, fuel rods are punctured to expose the spent fuel and allow O_2 to react with the UO_2 in an argon atmosphere at high temperature to form U_3O_8. The U_3O_8 expands, increasing the rupture of the cladding and pulverizing the fuel. H_2 gas is then used to reduce U_3O_8 back to UO_2. This oxidation-reduction cycle is repeated two times or more to achieve the desired particle size distribution. During the process, volatile fission products (Kr, Xe, and I) along with tritium are released and trapped, while the more refractory fission products (rare earths, lanthanides, and actinides) remain in the fuel. The resulting UO_2 fuel can be blended with enriched U or other recycled fuel, sintered into new fuel pins, and reclad into new fuel. During the sintering process, some semivolatile fission products, such as Cs and Ru, are also released (Majumdar et al., 1992). General Atomics is considering voloxidation (AIROX), or a variation of the process just described, as a part of an advanced fuel cycle in the future for recycling spent fuel from their Energy Multiplier Module (EM^2) gas-cooled fast reactor design. (Back and Schleicher, 2021).

Using pyroprocess separations technologies to partition spent nuclear fuel has several advantages (IAEA, 2021d). The nonaqueous medium used for pyroprocessing consists of inorganic salts, typically alkali (Li, Na, K) or alkaline earth (Mg, Ca, Sr, Ba) metal chloride or fluoride salts, which become liquid (molten) at relatively high temperatures (between 500 and 900°C). Where radiolysis-induced degradation of organics and water is a problem for aqueous processing in a high-radiation field, molten ionic salts have no structure in solution and thus do not decompose or otherwise degrade under high-radiation fields. This allows shorter cooling times between reactor discharge and reprocessing, as well as the potential to process higher-burnup spent nuclear fuel. Because these molten salts do not contain elements such as H or C that act as effective neutron moderators, the potential for critical accidents during processing is much reduced compared with aqueous processing. Furthermore, because molten salt systems, especially chloride-based systems, have a sufficiently high cross section for neutron absorption, they can tolerate feeds with higher fissile content compared with aqueous processes. This allows for more compact processing equipment to be used, which, in turn, leads to smaller footprints for facilities that use pyroprocessing compared with those that use aqueous separation technologies. An additional advantage of this smaller size is that the facility can more easily be collocated with reactors. Many MSR developers are likely to choose pyroprocessing for on-site recycling for this reason. Collocation also avoids some of the transportation issues for spent nuclear fuel, and existing physical protection systems and engineered safeguards can be leveraged across the entire plant site. From a nonproliferation viewpoint, collocation of spent nuclear fuel storage, reprocessing, and fabrication of recycled fuel—all within one protected facility—is thought to be a more effective safeguards strategy compared with performing partial separation of actinides and fission products (using, e.g., UREX and COEX™) at a centralized reprocessing facility.

Pyroprocessing also has disadvantages compared with aqueous reprocessing. A significant drawback for pyroelectrochemical processing is that it is inherently a batch process, which makes scale-up difficult. Lack of continuous processing capability limits the throughput of a process and, unless redundant or duplicative process equipment is in use, the overall process throughput will be hampered by single-point failures of the individual

[33] Actinide elements heavier than Pu (e.g., Am and Cm) do not form hexafluoride compounds.

[34] In processing, fluoride volatility is used in combination with other chemical/redox methods that recover the nonvolatile or less-volatile fluorides, such as adsorption/condensation or reductive extraction. Electrolysis, where the desired separations are achieved electrochemically, is an alternative method to halide volatility.

unit processes. Additionally, pyroelectrochemical processing lends itself to grouped recovery of the uranium and transuranic elements because of their similar redox potentials. As a result, the selectivity and product purity are relatively low (for fission products, separation factors are ~1,000 times smaller) compared with aqueous processes (IAEA, 2021d). Additionally, compared with the decades of experience with material accountancy and process monitoring for aqueous reprocessing, nuclear safeguards and security for pyroprocessing systems remain in their infancy (Coble et al., 2020). However, R&D based on electrochemical and spectroscopic techniques show promise for process monitoring and control, and for supplementing nuclear material accounting methods (Williamson, 2020). Lastly, to achieve the overall benefit of P&T using pyroprocessing technology, actinide losses to waste must be small (<0.1 percent). The committee was unable to find information on material recoveries and decontamination factors now being achieved by the EMT process, but past experience shows relatively poor and less efficient separations from pyroelectrochemical processing. In that case, a more extensive actinide drawdown operation by electrolysis or chemical reduction before passing the process salt on to waste form production steps would be needed to minimize actinide losses.[35]

4.3.6.5 Molten Salt Reactor: An Obvious Application of Pyroelectrometallurgical Technology

Liquid-fueled and liquid-cooled MSRs are unique in that the molten salt acts as both the fuel and the heat transfer fluid, and the reactors are adaptable to a wide range of fuel cycles (Hombourger et al., 2019). As mentioned in Chapter 3, MSRs can be designed to use an array of actinide fuels (thorium, uranium, plutonium) and operate over a wide range of neutron energies from thermal (with graphite used as a moderator) to fast (Holcomb et al., 2011). They can be operated as breeders (to produce startup fissile materials for other reactors), converters (no net production of fissile material), or burners (requiring periodic addition of fissile material); they can use a one-fluid system with the fissile and fertile materials in the same fluid or a two-fluid system—a more complex design with separate fluids for fissile and fertile materials. During MSR operation, the salt(s) flow(s) between the reactor core and heat exchanger(s) located external to the core where the heat is extracted and converted to steam for power production.[36] Some MSRs are designed to operate with fuel processing to remove fission products, while others run within a sealed environment where no fissile material is removed, and only denatured low-enriched uranium (by the addition of uranium-238) is added (LeBlanc, 2010). MSRs tend to be self-regulating because of the thermal expansion that causes the fission reaction rate to decrease with increasing temperature (e.g., large negative temperature coefficient of reactivity). However, MSRs pose a number of challenges, such as a higher-radiation-field environment leading to potential first-wall issues, more difficult maintenance and inspection operations, the use of unconventional methods of fissile material accountancy, and fewer barriers to radionuclide release. On the other hand, MSRs require no fuel fabrication in the traditional sense, as required by solid fueled reactors; avoid in-core materials limitations, such as that exhibited by cladding at high burnup; and have minimal excess reactivity, the ability to breed, and the potential to burn transuranic elements (Holcomb, 2015).

The choice of the fuel salt for an MSR is constrained by several factors, some of which include (1) relatively low melting temperature (typically around 500°C or below); (2) for thermal MSRs, a low capture cross section; (3) for fast MSRs, a low scattering cross section; and (4) a relatively high solubility for actinides. Since actinide elements are known to form chemically and radiolytically stable salts with the halides, and halide salts have an acceptably low parasitic neutron absorption, they are good candidates for use as both fuel salts and coolants in MSRs. As a result, much of the early work on MSRs (e.g., the Molten Salt Reactor Experiment at Oak Ridge National Laboratory) employed halide salts and in particular, fluoride salts, which has led to a more extensive literature available for fluoride than for chlorides salts. However, different properties can be exploited depending on the specific MSR design. Whereas the use of fluoride salts results in softer neutron spectra, the heavier halide (chloride) provides a harder neutron spectrum, which tends to favor breeding and burning of minor actinides. MSRs for actinide burning will favor the use of molten chloride salts because of their higher actinide solubilities.

[35] This sentence was altered after release of a prepublication version of the report to clarify treatments needed before processing.

[36] Several options are being explored in addition to steam turbine generation of electricity via the conventional Rankine cycle, including advanced heat exchangers using other heat transfer fluids such as secondary molten salts or a gas such as helium (Brayton cycle gas turbine).

There are other constraints on the choice and composition of fuel salts. For molten salt chloride fast reactors, enrichment of chlorine-37 is needed not only to reduce neutron absorption and enhance breeding but also to limit the production of the long-lived (3×10^5 years) chlorine-36 isotope, which is problematic given that chlorine is readily soluble in water. Currently, chlorine-37 enrichment is not performed at commercial scale anywhere in the world (Napier, 2020). On the other hand, tritium production and containment are important issues for thermal MSRs that use fluoride salts containing lithium, such as $2LiF\text{-}BeF_2$ (FLiBe), since at elevated temperatures, tritium can permeate structural alloys (Holcomb, 2017). Neutron reactions on lithium-6 (which is 7.6 percent abundant) within the fuel salt generate copious amounts of tritium (1 Ci tritium/MWth [megawatts thermal]/day), some of which would be released to the cover gas during normal operations (McFarlane et al., 2020; Sorensen, 2021). By using enriched ^7LiF (≥99.99 percent), the neutron absorption problem can be mitigated to a large extent. (See Box 4.2 for details on current industrial uses of lithium-7.)

Fission products in MSRs pose a challenging problem in that, depending on their half-lives (some of which are very short), as well as their chemistry and volatility, they can be found almost anywhere within the reactor volume and require different treatment methods. Volatile noble gas fission products (krypton and xenon) will escape the salt and end up within the reactor cover gas. Since many fission products have krypton or xenon precursors (decay daughters cesium, barium, rubidium, and strontium), a significant fraction of cesium, strontium, and even iodine can end up in the off-gas, so managing the volatile effluents from an MSR is an important design consideration. Off-gas treatments, such as those involving in situ helium sparging, can be used to separate, capture, and store (for decay) the volatile fission products (e.g., continuous removal of krypton and xenon fission products) (Riley et al., 2019). The more noble metal fission products, which plate out as metals on reactor components and end up as particulates in the salt, can be filtered out physically, as they tend to come to the surface during sparging. Most of the fission products, as well as the actinides (including the lanthanide fission products), remain dissolved in the salt. For MSRs that require processing the fuel salt to remove fission products (e.g., MSRs breeders using thorium-232), a fraction of the salt can be removed and processed online or in batches to remove fission products and, if breeding thorium to uranium-233, to isolate protactinium.

The fuel processing operations are expected to be carried out in what amounts to a chemical processing facility (CPF) directly attached to the reactor (Fredrickson et al., 2018). The CPF is appropriately sized to support reactor operations and serves both as a reprocessing facility for MSR fuel and a means for managing the chemistry of the reactor's molten salt. Performance of the reactor and the CPF are intimately and complexly coupled for MSRs, so a key function of the CPF is to ensure that the properties of the salt are appropriate for optimal reactor performance. Salt processing is important for controlling such impurities as oxygen and water in the salt, as well as the redox potential of the salt, to limit corrosion (Frederickson et al., 2018). Furthermore, fuel salt chemistry in MSRs is complex, and its composition (e.g., quantities of fissile and fertile isotopes; fission, transmutation, radiolysis, and corrosion products) is continually changing with time. Actinide management in MSRs can take a variety of forms depending on the reactor design (Forsberg, 2007; Forsberg and Greenspan, 2003). Theoretically, the actinides could remain in the fuel and be burned (as planned for Terrestrial Energy's IMSR-400 with no salt processing or the net breed-and-burn molten chloride fast rector being developed by TerraPower), or they could be isolated in the CPF and reintroduced later as fuel (as planned by developers of ThorCon's thermal thorium-converter reactor, Flibe's thermal liquid-fluoride breeder reactor, and Moltex's stable salt-waste burner reactor) (Delpech et al., 2009; Holcomb et al., 2011; Hombourger et al., 2019). As mentioned in Section 3.2.3.5, a 2021 EPRI report noted that salt lifetimes in fast chloride MSRs may be limited by the accumulation of fission products in liquid fuel molten salts over time unless steps are taken to clean up the fuel salts of fission products (EPRI, 2021a). For example, fuel salt processing would be useful for removing parasitic neutron-absorbing fission products, maintaining the desired salt chemistry, managing corrosion potentials, and ensuring the actinides remain soluble.

The wide range of MSR design concepts each has numerous variants with their own potential advantages and challenges. Basic elements of fuel cycles that might support MSRs have been identified, and some technologies have been demonstrated to some level of success. However, it is too soon in the development of MSRs to discuss and analyze their associated fuel cycles. If and when MSR designs become more mature, their associated fuel cycles will become more obvious, as they will depend on the objectives of reactor operation (e.g., power production, breeder with online salt processing, actinide burning).

BOX 4.2
Current Industrial Uses of Lithium-7

Currently, ^{7}LiOH (lithium hydroxide) is being used in the coolant systems of pressurized water reactors (PWRs) both to stabilize the pH of the coolant to reduce corrosion and to remove coolant contaminants using a demineralizer (GAO, 2013a). Lithium-7 has not been produced in the United States since 1963 when the COLEX[a] facility was shut down because of environmental and health concerns presented by the industrial-scale use of mercury. As a result, the nuclear power industry now relies on Russia and China to supply the lithium-7 needed to support its PWR fleet (the annual requirement for the roughly 65 PWRs in the United States alone is ~300 kg) (WNA, 2017b). The Russian Novosibirsk Chemical Concentrates Plant (NCCP) in Siberia supplies about 80 percent of the world's production of lithium-7 (~1 MT of lithium-7 hydroxide monohydrate per year with a purity of up to 99.95 percent), with the remainder coming from Shanghai Institute of Applied Physics (SINAP) in China. NCCP uses electrolysis of lithium chloride using a mercury cathode as the enrichment technology, while SINAP uses centrifugal extraction along with countercurrent extraction (Napier, 2020).

The question of whether supplies from Russia and China could reliably meet the domestic demand for lithium-7 was addressed by the U.S. Government Accountability Office (GAO) in a 2013 report. The question was timely, especially in light of the potential impact of 25 new PWRs expected to come online by 2015 and the pursuit of technologies for both high-temperature fluoride salt–cooled reactors and molten salt reactors (each requiring thousands of kilograms of lithium-7 for operation) by 2020 in China (GAO, 2013a). Although Russia's PWR fleet does not rely on lithium-7, it was noted that "in June 2014, NCCP signed a three-year contract for supply of lithium-7 of higher purity > 99.99% purity to China" (WNA, 2017b). Due to the increased demand from China, it has been suggested that Russia's supply of lithium-7 may not be available to meet the U.S. demand. The situation may be further exacerbated by the ongoing war in Ukraine. The GAO found that reliance on "two foreign producers to supply a chemical that is critical to the safe operation of most of the commercial nuclear power reactors in the United States places their ability to continue to provide electricity at some risk" and recommended that the "Secretary of Energy direct the Isotope Program, consistent with the program's mission to manage isotopes in short supply, to take on the stewardship role for lithium-7" (GAO, 2013a). Three options were identified to reduce near- and long-terms risks that included "building a domestic reserve of lithium-7, building domestic capability to produce lithium-7, and reducing pressurized water reactors' reliance on lithium-7" (GAO, 2013a). The GAO noted that the Isotope Program within the U.S. Department of Energy funds research and development associated with alternative methods for lithium isotope separation. Some of these include processes based on electromigration and the use of crown ethers on solid supports such as resins or in organic solvents where the separation is achieved by solvent extraction between the organic phase and aqueous phase containing lithium hydroxide (McFarlane et al., 2019). At this time, there has been no industrial-scale demonstration of any of these alternative separation methods for lithium-7 (Holcomb, 2017).

[a] In the 1950s and 60s, lithium-7 was produced as a by-product of enriching lithium-6 for the United States' nuclear weapons program using the COLEX process, a column extraction process that used an aqueous solution containing lithium hydroxide and a mercury amalgam in a countercurrent extraction process (Napier, 2020).

4.3.7 Analyzing the Main Motivations for LWR-Inspired Advanced Fuel Cycles

Historically, the two main drivers for developing advanced fuel cycles that build on LWR technology have been better utilization of natural resources and, at a later time, reduction in the long-term radiotoxicity of the resulting waste, which comes primarily from contained transuranic elements (TRU) (see Box 4.3 on radiotoxicity).

Table 4.3, adapted from EPRI (2010b), shows several potential fuel cycles and their expected impact on natural uranium consumption and on the mass of TRU waste being eventually disposed of in a geologic repository. Five fuel cycles were selected for illustration in Table 4.3:

- LWR—Once-through fuel cycle (OTC): uranium dioxide reference cycle
- LWR—Monorecycling of plutonium in the form of MOX, followed by disposal of the spent MOX fuel in a geologic repository
- LWR—Multirecycling of plutonium in the form of MOX fuel with enriched uranium (MOX-EU)
- LWR + FR (fast reactor)—Multirecycling of plutonium in an FR in the form of MOX-FR, with partitioning of americium and curium, once-through transmutation of americium, and 100-year storage of curium
- FR—Multirecycling of plutonium in the form of MOX-FR with partitioning and homogeneous transmutation of transuranic elements

These fuel cycles were selected first on the basis of their improved plutonium management in LWRs and second on their management of minor actinides (neptunium, americium, and curium) by introducing fast reactors and integrating partitioning and transmutation technologies in the fuel cycle.

The analyses were performed using a steady-state approach, that is assuming that all reactors operate at constant power and all mass flows have reached an equilibrium. Importantly, the equilibrium phase has to be preceded by a deployment phase. The deployment phase ends when the heavy metal inventory of the fuel cycle has built up to the point where the fuel composition has reached equilibrium. The deployment phase lasts many decades for a transmutation strategy using multirecycling (National Research Council, 1996; NEA-OECD, 2006b).

Natural uranium consumption: A fleet of all PWRs (100 percent) operating in a once-through fuel cycle is used as a reference and assigned the relative value of 1 for natural U consumption. Assuming the same amount of electrical energy generation, the natural U consumption rate of the other fuel cycles is a fraction of the reference PWR fleet value and varies from slightly lower than 1 for various recycles of Pu in PWRs to <0.01 for a fleet of fast reactors operating in the breeder mode in which all TRUs are continuously recycled. Calculated values for such a cycle yield a natural U consumption of 0.036 with no U recycle and 0.004 with U recycle (NEA-OECD, 2006b). Table 4.3 illustrates that fast reactors are required for extending the use of natural U resources by a significant factor.

TRU content going to the repository: This exercise assumes that the TRU masses going to the repository are the result of a 1-GWe plant operating at 100 percent capacity for 1 year. For the once-through cycle, all TRUs are from spent UOX fuel. For the monorecycle option, TRUs are from reprocessing (~15 percent) and spent MOX fuel (~85 percent). For the multirecycle option of Pu in PWRs, TRUs are from reprocessing. For the scheme involving fast reactors and partitioning and transmutation, most of the TRUs are coming from reprocessing losses.

Table 4.3 shows that the waste management benefits of Pu monorecycling in LWRs, when assuming disposal of the spent MOX fuel in a geologic repository, are limited. Most of the waste management benefits accrue over time when fast reactors are part of the fleet of nuclear power plants.

TRU cycle inventory: Also shown in Table 4.3 are the in-process and in-reactor TRU inventory in the equilibrium fuel cycle at any given time, assuming a single 1-GWe plant operating at 100 percent capacity. Recycling of fuels leads to high in-pile and out-of-pile TRU inventories.

There is an inverse correlation between the TRU mass flow going into the geologic repository and the TRU mass in the fuel cycle. In other words, positive benefits for waste management become negative attributes when reactor operations (licensing challenges), fuel fabrication (remote operation versus glove-box operation), and reprocessing (TRU partitioning, radiological protection due to increase in TRUs, criticality safety) are considered. Closing the fuel cycle with partitioning and transmutation of the TRUs in fast reactors eliminates, with the exception of process losses, their disposal in a geologic repository. However, it is done at the expense of handling, irradiating, and storing large TRU inventories in close proximity of places where people live (public acceptance issues). As further discussed in Chapter 5, if all TRU is directly disposed of in a geologic repository, the majority is not expected to make it to the biosphere because of multiple barriers—all of which allow time for decay and retard or prevent dispersion.

TABLE 4.3 Comparison of (1) Natural Uranium Consumption Compared with the Once-Through Cycle; (2) TRU Mass Going into a Geologic Repository; (3) TRU Mass in the Equilibrium Fuel Cycle; and (4) Requirements for Advanced Reactors and Facilities for Five Different Fuel Cycle Options[a]

Nuclide	LWR Once-Through Cycle (OTC)	LWR Monorecycling of Pu	LWR Multirecycling of Pu	LWR + FR Multi- (Pu/Np) & Mono- (Am/Cm) Recycling	FR Multi (Pu/MA) Recycling
Reactor Fleet	PWR (100%)	PWR (100%)	PWR (100%)	PWR (44%) and FR (56%)	FR (100%)
Natural U consumption (compared with OTC)					
	1	0.89	0.87	0.44	<0.01
TRU Content Going to Repository Assuming 0.1% Loss in Separation Processes [kg/year]					
Pu	230	153	0.37	2.10	1.25
Np	16.2	16.6	14.4	0.02	0.0066
Am	6.35	16.2	39.4	0.35	0.055
Cm	3.3	8.11	19.7	2.06	0.013
Total TRU	256	194	74	4.53	1.32
TRU Cycle Inventory (reactor + fabrication + reprocessing) [kg]					
Pu	767	3,285	4,818	10,293	17,520
Np	53	131	116	241	88
Am	22	88	307	438	701
Cm	11	44	158	263	175
Total TRU	853	3,548	5,399	11,235	18,484
Requirements for Advanced Reactors and Facilities					
	None required	None required	ALWRs[b] and reprocessing[c]	Fast reactors and advanced FBR fuel reprocessing	Fast reactors and advanced FBR fuel reprocessing

[a] Values in this table were derived from the information in (NEA-OECD, 2006b); values were calculated by the French Alternative Energies and Atomic Energy Commission (CEA) and documented in "Synthèse des Résultats des Recherches sur l'Axe" 1, 2005. Small differences between the NEA and CEA reports are due to the different assumed values for some of the input parameters, such as reactor efficiency, burnup, and storage times. For all fuel cycles involving LWRs, the same burnup of 60 GWd/MTHM and efficiency of 34 percent were assumed. In all cases, reprocessing losses sent to waste were assumed to be 0.1 percent.

[b] Advanced LWRs, such as the third-generation European Pressurized Water Reactors (EPRs), licensed for handling fuel with high Pu content, and/or LWRs with a higher moderator-to-fuel ratio compared with existing LWRs.

[c] Compared with the current situation in France, where only low-enriched U fuels are recycled, new facilities may be required for processing Pu-rich spent fuel at the required industrial throughput.

NOTE: ALWR = advanced light water reactor; FBR = fast breeder reactor; FR = fast reactor; LWR = light water reactor; MA = minor actinides; PWR = pressurized water reactor; TRU = transuranic element.

SOURCE: Adapted from EPRI (2010b).

Requirements for advanced reactors and facilities: The need for advanced, not-yet-available facilities required for fuel cycle operation is an indication of technological challenges and a potential detriment to economic competitiveness. As stated in Chapter 2, the U.S. Energy Information Administration notes that renewable energy incentives and falling technology costs support robust competition in the electricity mix (EIA, 2021b). Advanced nuclear technologies will have to prove that they can be cost competitive for providing energy in a carbon-constrained world (MIT, 2018).

4.3.8 Long Timescales to Implement LWR-Inspired Advanced Fuel Cycles

The establishment of an equilibrium fuel cycle for advanced schemes involving multirecycling can take a long time. For example, the 1996 National Research Council report on separation and transmutation showed that some advanced fuel cycles require several centuries of sustained operation before achieving the very low transuranic

mass rates in a repository (National Research Council, 1996).[37] In addition, an MIT analysis of alternative fuel cycles for nuclear power growth scenarios through 2100 concluded that *fuel cycle transitions take 50 to 100 years* [emphasis added] (MIT, 2011). Also importantly, the simplest form of the existing LWR fuel cycle (i.e., the once-through fuel cycle) is still incomplete in the United States, as there is no geologic repository. Its completion is expected in Finland, Sweden, and France during this decade or the next; thus, it will have taken about 60 to 75 years, in the best cases, to complete the once-through LWR fuel cycle. In the United States and several other countries, completion of the once-through LWR fuel cycle will likely require more than a century. Based on the French experience, implementing the LWR-inspired fuel cycle involving fast reactors fueled with plutonium, if it is indeed pursued to completion, cannot be expected prior to the end of this century. Any fuel cycle involving processes that require administrative controls and public/policy support for time periods longer than several decades will be challenging to implement.

4.3.9 Implications of Using Multirecycling with the P&T Strategy and Fast Reactors to Reduce Transuranic Wastes and Repository Heat Load

Does multirecycling using P&T offer advantages for waste repositories? Notably, only transmutation strategies with fully closed fuel cycles can reduce transuranic (TRU) waste by 100-fold. Partially closed fuel cycles, such as the multirecycling of Pu in LWRs, are easier to implement but cannot achieve high TRU reductions. The other potential advantages for multirecycle P&T considered here are (1) decreasing the high-level waste burden in terms of the inventory of long-lived TRU radionuclides as well as the time required to safely sequester the waste and (2) reducing the heat load of the wastes. Whereas closing the fuel cycle makes recycling fuel (i.e., reprocessing and fuel fabrication) much more difficult, removing the actinides, which includes uranium, from the waste stream makes handling and managing of the waste in a repository somewhat easier and less complicated, because only about 3 percent of the mass of the spent nuclear fuel, consisting of the fission products, goes to the repository.

4.3.9.1 Radiotoxicity Reduction

Reducing both the volumes and radiotoxicity and shortening the time required for waste storage are the primary current motivations for developing P&T scenarios. Discussions of the impact of the radiotoxicity of actinides in radioactive waste for various P&T scenarios are commonly found in the literature. Often, the radiotoxicity of waste from spent fuel is compared with that of uranium ore as a reference point and called "relative radiotoxicity" to avoid the use of the dose unit. The comparison to uranium ore, as pointed out by Piet (2013), is not a regulatory concept but is used for reasons such as (1) the concept is easier to explain to a nontechnical audience; (2) it seeks to compare the hazard as greater or lesser than natural uranium existing in the environment[38]; and (3) useful comparisons can be made if no site-specific dose assessments are available. However, some of the radiotoxicity calculations are incorrect because they do not include the decay products of all isotopes in the waste that either captured neutrons or did not fission while in the reactor. See Box 4.3 for details on radiotoxicity's use and applications.

Magill et al. (2003) reported on the impact of P&T scenarios on the radiotoxicity of actinides in radioactive waste. In this work, the efficiency of P&T was calculated for an initial inventory of actinides (for fuel with 4.2 percent enrichment, 50 GWd/MT burnup, and 6 years cooling time) in grams per ton of spent PWR fuel through

[37] See for example, Figure 4-1 in National Research Council (1996), which plots the "transuranic (TRU) ratio," y(t), as a function of time of transmutation operation. The TRU ratio, y(t), is defined as the ratio of the total inventory of transuranics sent to waste disposal as a function of time "for the reference once-through LWR fuel cycle (no fuel reprocessing, no recycle, and no transmutation) to the total inventory of TRUs at time t in the transmuter itself (i.e., fast reactor), in its fuel cycle, and in process wastes" (National Research Council, 1996). Assuming an advanced fuel cycle based on fast reactors (advanced light water reactors) operating "at constant power for 100 years and then terminated," the TRU ratio would reach "a value of 6.9. Reducing the TRU inventory by only a factor of 6.9 below that of the reference once-through fuel cycle is far from the goals proposed for transmutation" (National Research Council, 1996). Based on a linear extrapolation of the plot in Figure 4-1, achieving y(t) = 20 or 70 would require ~300 years or 1,000 years, respectively, based on a fast reactor fuel cycle operating at constant power.

[38] As pointed out by Piet (2013), "there is a sense (sometimes implicit) that if one can be adequately protected against natural ore hazards, then it is possible to be adequately protected against waste with comparable (or lower) radiotoxicity."

BOX 4.3
Radiotoxicity: Use and Application

Radiotoxicity is a measure of the dose (Sv) unique to a specific radionuclide. Despite being expressed in units of dose, radiotoxicity is not a measure of the potential dose because it does not consider the amounts of the radioactive inventory that are released into the environment that then result in an exposure to human beings. Radiotoxicity does not consider the effectiveness of any barriers to release, transport processes, such as sorption, or dilution of radionuclides.

The radiotoxicity of a specific radionuclide depends on its specific activity and the type of radiation emitted during decay, as well as how that radionuclide is transported through or finally comes to rest in the body. The behavior of radionuclides within the body is represented by a dose conversion factor (dcf) that is unique to each radionuclide and the mechanism of exposure. Mathematically, radiotoxicity can be calculated by

$$\text{radiotoxicity (Sv)} = \text{I (radionuclide inventory in kg)} \times \text{dcf (Sv/kg)}$$

If the inventory is a mixture of radionuclides, the radiotoxicity is the sum of the weighted mass fractions of each radionuclide.

An important distinction must be made between external and internal exposures. An external exposure is by irradiation from a source outside of the body, such as "ground shine" as found at a contaminated site, or "cloud shine" that results from being immersed in contamination, such as fallout from a nuclear detonation. Internal exposure is from radioactive substances that are ingested (swallowed) or inhaled into the lungs. The dose conversion factors will differ depending on the type of exposure (internal versus external); similarly, the dose conversion factors differ between ingestion and inhalation. As an example, external irradiation by actinides on the skin is low because the decay is via alpha-particles that have low penetrating power, tens of microns, and do not pass through the skin. In contrast, inhalation of actinides that become lodged in the recesses of lung tissue have an unshielded effect on surrounding tissue. *For applications to radioactive waste management and disposal, ingestion is appropriate for radionuclides transported and consumed from groundwater, while inhalation is more appropriate for certain accident scenarios that model the release of radionuclides into the air.*

The convenience of the radiotoxicity calculation for nuclear materials of complex compositions, such as spent fuel, is that for any specific time and composition, the radiotoxicity is simply the weighted sum of all the radionuclides in the material corrected for the loss caused by radioactive decay over time. Because different radionuclides have different half-lives, the proportions of the radionuclides change with time. Short-lived radionuclides disappear, and longer-lived radionuclides, such as ^{238}U, ^{239}Pu, ^{237}Np, and ^{129}I, persist and become the major contributors to radiotoxicity. Additionally, one must consider the radiotoxicity of the decay products. Thus, although some fuel cycles lower the actinide inventory, this does not necessarily reduce the long-term hazard.

all successive burnups, reprocessing and recycling, and sums all of the reprocessing losses at each step. These data were used along with effective dose coefficients to calculate total actinide ingestion radiotoxicity as a function of time by three independent research groups: ITU and FZK in Germany, and CEA in France.[39] The agreement among the groups on the actinide mass inventories was better than ±10 percent for U and Pu, better than ±20 percent for Am isotopes, ±25 percent for the lighter Cm isotopes (242 and 244), and ±70 percent for the heavier Cm isotopes (245 and 246). The groups agreed on radiotoxicity with time, with small differences unnoticeable on a logarithmic plot.

Figure 4.12 shows these results for the evolution of ingestion radiotoxicity in a geologic repository as a function of time for several cases.

[39] ITU, Institute for Transuranium Elements, Karlsruhe, Germany, FZK. Forschungszentrum, Karlsruhe, Germany, and CEA-Cadarache, DER/SPRC/LEPh, France.

Commonly, the impact of different strategies for nuclear cycles are evaluated by calculating the changes in the radiotoxicity with variations in fuel cycle strategy (e.g., the impact of reprocessing or different geologic repository types). Typically, the radiotoxicity of spent nuclear fuel is calculated for inhalation, which is an unlikely but conservative scenario. However, extreme scenarios, such as ingestion of highly radioactive spent nuclear fuel, can lead to misleading interpretations unless more appropriate parameters are also considered, such as mechanisms of release and the mobility of radionuclides in the environment. Thus, the calculation of the changing composition of spent nuclear fuel and its radiotoxicity on inhalation provides a single, qualitative measure of the risk posed over time by the spent nuclear fuel source term.

In evaluating the impact of different fuel cycle strategies, radiotoxicity is only one of several parameters that can be assessed. In the case of evaluating the performance of a geologic repository, the accessibility of radionuclides to the environment is paramount. The accessibility depends on the performance of waste packages, the solubility and speciation of radionuclides, the transport mechanism, and sorption onto surrounding mineral surfaces in the host rock, to name a few. *There is no simple or direct relationship between radiotoxicity of spent fuel that is inhaled, an extreme scenario, and the dose estimates of the performance assessment of a geologic repository.*

Two points summarize this discussion:

1. Mathematically, radiotoxicity is defined as the inventory of radioisotopes (in mass or activity units) weighted by an appropriate dose conversion factor. If the inventory is a mixture of radionuclides, the radiotoxicity is the sum of the weighted fractions of each radionuclide. Although not strictly correct because radiotoxicity varies with time, *radiotoxicity* has become synonymous with describing the "harm," "potential danger," or "hazards" if people are exposed to a given inventory of radionuclides (Hedin, 1997; Magill et al., 2003; Paunov and Naydenov, 2020). At a given point in time or time period, radiotoxicity can be used to compare the relative hazards of inventories of different radioactive materials if it is calculated correctly and includes all of the decay products (Piet, 2013). Risk, on the other hand, is the product of radiotoxicity (hazard) and accessibility (exposure) (Hedin, 1997).
2. Radiotoxicity alone is a poor metric for repository performance and risk to the public from waste disposal because it does not take into account mechanisms that exist naturally or are put in place, such as geologic conditions, engineered barriers, or waste forms, to significantly limit or prevent radionuclide accessibility (e.g., exposure due to radionuclide release and migration through the environment). Likewise, radiotoxicity alone is not a good metric for judging benefits or comparing impacts of advanced fuel cycles on near-term worker dose and public health risks and/or on long-term disposal system performance (Apted, 2012).

SOURCES: For a more detailed discussion, see Apted et al. (2012) and Piet (2013).

As seen from Figure 4.12, the radiotoxicity of spent nuclear fuel in a geologic repository will remain above the reference level for (1) 130,000 years for the once-through fuel cycle; (2) between 500 and 1,500 years for high-level waste after P&T, depending on assumptions regarding P&T efficiencies; and (3) 270 years when all actinides are removed in a fully implemented P&T scenario in which only fission products are sent for geologic disposal. For case (3), 270 years represents a lower limit, as process losses will inevitably occur throughout the fuel cycle; therefore, no fuel cycle can be considered perfectly closed. This exercise is meant to demonstrate that if P&T could be implemented at the required efficiencies, it could have an impact on the time required to safely sequester the high-level waste in a geologic repository. Magill et al. (2003) caution that "the reprocessing of spent nuclear fuel inevitably results in the production of secondary waste" (e.g., contaminated resins, fuel hulls and end pieces), and the management of such waste and its impact on geologic disposal require further detailed study.

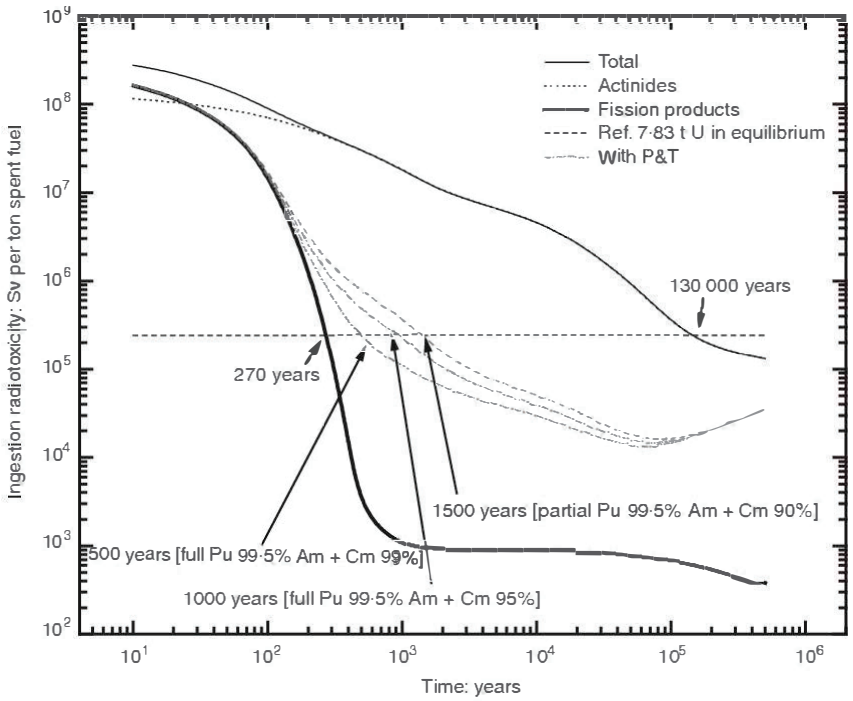


FIGURE 4.12 Evolution of ingestion radiotoxicity as a function of time for an initial inventory of actinides for fuel with 4.2 percent enrichment, 50 GWd/t burnup, and 6 years cooling time. The radiotoxicity reference level (horizontal dashed line) is set at 7.83 MT of natural U, or the amount of natural U required to produce 1 MT of fresh fuel enriched to 4.2 percent ^{235}U.
NOTE: P&T = partitioning and transmutation.
SOURCE: Magill et al. (2003). Courtesy of the Nuclear Institute.

As pointed out in Box 4.3, radiotoxicity is only one of several parameters that can be used to evaluate the impact of different fuel cycle scenarios on repository performance. Any discussion of radiotoxicity regarding waste only considers the consequences of release of radioactive material in the absence of a repository. Chapter 5 discusses how the design of a repository, including the choice of the geologic setting and engineered barriers, can significantly mitigate the potential release and migration of certain elements away from the repository.

The actual hazard posed is a stronger function of the mobility of the radionuclides in and around the geologic repository than the radiotoxic inventory of the radionuclides in the repository. For example, the radiotoxicity due to the actinides when in a repository with reducing conditions is minimal since their mobility in the biosphere is much less compared with that of some long-lived fission products that exist as anions (Grambow, 2008). Reducing the minor actinide inventory in a chemically reducing geologic repository is therefore only important in the case of the scenario of inadvertent or accidental intrusion.

4.3.9.2 Heat Load Reduction

Fully closing the fuel cycle using fast reactors—that is, by multirecycling of Pu and P&T of the minor actinides—can reduce the heavy metal mass by more than three orders of magnitude compared with the once-through fuel cycle. But what is the impact of the heat load on the repository if the actinides are removed? The residual heat power associated with spent nuclear fuel follows essentially the same trend as radioactive decay, being driven by short-lived fission products for the first 50–70 years and then by the alpha-emitting actinides (Pu, then minor actinides) at later times. Because of their decay characteristics, Pu and the minor actinides represent the largest contributor to the heat load in a geologic repository, particularly at longer times (>100 years). Although the total residual heat from the actinides in spent nuclear fuel is higher than that from the decay of fission products,

there is little if any temperature rise expected in a repository at later times because of the long-half lives of the actinides; their rate of heat release is much lower and released over a much longer time period (>100,000 years).

In contrast, the relatively short half-lives of fission products provide a significant thermal pulse and temperature rise in the repository early on (from emplacement to roughly 300–400 years, peaking around 40–50 years) as their heat is released over a much shorter period (a few hundred years). It is the thermal load along with the thermal conductivity of the geologic setting that determines the temperature rise in a repository. Temperature increases may be able to be managed by appropriate spacing of spent nuclear fuel or high-level waste canisters, but the amount of decay heat per disposal canister will be limited depending on the backfill and the local geologic environment of the repository.

Because the repository temperature (e.g., heat load), not the waste volume, is a key factor in repository design, a properly designed repository will have a disposal density (canister spacing) that manages the thermal pulse from the fission products from spent nuclear fuel, and doing so should be well within a conservative envelope to handle the late thermal load from the actinides. As mentioned previously, recycled fuel, such as spent mixed oxide fuel, has a higher heat content than spent uranium oxide fuel. Direct disposal of spent mixed oxide could largely negate any benefit from reprocessing with regard to decreasing decay heat per unit electrical energy generated and incur costs for long-term storage of spent mixed oxide outside of a repository (CBO, 2007). Factors that could influence the total costs of this option are the costs of disposing the low-level wastes, including GTCC, generated from reprocessing and any engineering or management strategies deployed to handle the higher heat load.

4.3.10 Observations on the Back End of the Fuel Cycle

When considering advanced fuel cycle options, the question to be asked is, What has changed in the United States that requires a reevaluation of the decision not to reprocess spent fuel?

In 1996, at the request of DOE, the National Academies convened a committee of experts to evaluate state-of-the-art separations technology and transmutation (S&T) systems (National Research Council, 1996). As introduced in Chapter 1, the principal recommendations from that committee were as follows:

- "None of the S&T system concepts reviewed eliminates the need for a geologic repository. DOE should continue efforts to develop a geologic repository for spent LWR fuel.
- The current policy of using the once-through fuel cycle for commercial reactors, with disposal of the spent fuel as HLW, should be continued.
- Fuel retrievability should be extended to a reasonable time (on the order of 100 years) to avoid foreclosing alternative fuel strategies that may be in the national interest.
- Research and development should be conducted on selected topics to support the cost-effective future application of S&T of commercial spent fuel and separations for defense waste applications." (National Research Council, 1996)

The report went on to say:

> A sustained, but modest, and carefully focused program of research and development over the next decade could prepare the technical basis for advanced separation technology for the radionuclides in spent LWR fuel and for decisions on the possible applications of S&T as part of the more efficient future use of fissionable resources. The research and development effort should focus on the factors that strongly influence fuel-cycle economics, especially the costs of reprocessing spent LWR fuel, minimalization of long-lived radionuclides to secondary wastes in the reprocessing cycle, and on the need to minimize the possible increase in proliferation risks that could result from the commercial use of plutonium in recycle fuels. (National Research Council, 1996)

The report also found that "The construction and operation of an S&T system would require, in addition to several new types of facilities, the resolution of major institutional, public policy, and public acceptance issues" (National Research Council, 1996). It further concluded that, should these issues be overcome, "An S&T system

of appropriate scale must be operated for many decades to achieve the permanent benefits to the repository and other parts of the nuclear fuel cycle" (National Research Council, 1996).

The report also concluded that implementing P&T to reduce substantially the amount of long-lived minor actinides that would be placed in a geologic repository necessitates a commitment to this approach for more than a century. Such an effort would require "a cohesive national intent and commitment" on the part of the U.S. government, else S&T operations could be shut down early and "facilities related to recycling (i.e., reprocessing and fuel fabrication plants) are likely to be a loss" (National Research Council, 1996).

It has now been 26 years since that report was issued, but its conclusions remain even more valid now than before, given the diminishing contributions of LWR technology to electricity generation in the United States. The availability and price of uranium have remained stable and relatively inexpensive, so there is no compelling reason at this time to deploy reprocessing technologies and advanced reactors on the sole basis of economic considerations (NEA and IAEA, 2020).[40] The once-through fuel cycle remains the safest and most cost-effective option in the short term. However, this option hinges on the siting, construction, and operation of a geologic repository.

4.4 COST ESTIMATION OF DIFFERENT FUEL CYCLE OPTIONS

The committee was tasked with examining the potential costs of the different nuclear fuel cycles required for advanced nuclear reactors that could be commercially deployed by 2050. A number of published reports and papers have addressed the economics of nuclear power, including fuel cycles (e.g., Black and Peterson, 2018; Black et al., 2019; Bunn et al., 2003; EPRI, 2007, 2009b, 2010a,c,d; Huff, 2019; MIT, 2011; NEA-OECD, 1994, 2006b, 2013; Recktenwald and Deinert, 2012; Rothwell et al., 2014; Schneider et al., 2009; Shropshire et al., 2021).

In the 2011 MIT study "The Future of the Nuclear Fuel Cycle," three fuel cycles were modeled in detail: (1) once-through LWR fuel cycle; (2) monorecycling of Pu in LWRs with direct disposal of recycled MOX spent nuclear fuel; and (3) a closed fuel cycle using reprocessing where spent uranium and TRU are recycled back to the fast reactors operating with different conversion ratios (CRs). These fast reactors and their CRs are an actinide burner (CR <1), a self-sustaining converter with CR = 1, and a breeder reactor with a CR >1 with excess TRU used to start additional fast reactors. For that study, the levelized cost of electricity (LCOE) was the measure of cost, since LCOE is the "constant price that would have to be charged in order to recover all of the costs expended to produce the electricity, including a return on capital" (MIT, 2011). LCOE was divided among the following main components: cost of the front-end fuel cycle (cost of raw U and conversion, enrichment, and fuel fabrication); reactor capital costs and nonfuel operating and maintenance (O&M) costs; and the cost of the back end of the fuel cycle.[41] The MIT (2011) study authors assessed:

- For the once-through cycle, the reactor capital costs (81 percent) and the reactor (non-fuel) O&M costs (9 percent) dominate the LCOE with front-end and the back-end costs coming in at 8 percent and 1.6 percent, respectively. For closed fuel cycles with fast reactors, the reactor capital and O&M are expected to be an even higher portion of the LCOE.
- The "most important conclusion from a comparison of the LCOE across the three fuel cycles is the differences between them are small relative to the total cost of electricity" with the highest being the fast reactor cycle and the lowest the once-through cycle (with less than 3 percent range from the highest to the lowest).

[40] Information on fissile content in the discharged fuels from advanced reactors using HALEU was not provided to the committee, so consideration of recovery and recycling of uranium-235 would be only speculative at this time. Most advanced reactor designers are opting for a once-through cycle as their initial approach.

[41] For the once-through fuel cycle, the back-end fuel costs included above-ground interim spent nuclear fuel storage, transportation, and the cost of disposal in a geologic repository. For the monorecycling fuel cycle, reprocessing, transportation, mixed oxide fuel fabrication, and high-level waste disposal costs were added, along with positive and negative credits for such things as fuel utilization and increased costs of spent mixed oxide disposal. For the closed fuel cycle, higher costs for reprocessing, fuel fabrication, and high-level waste disposal were added, along with similar positive and negative credits. In the case of the closed fuel cycle, a higher capital and O&M cost for fast reactors relative to LWRs was included.

- "The benefits to resource extension and waste management of mono-recycling in LWRs using mixed oxide fuel, as is being done in some countries, are minimal."
- "A conversion ratio near unity is acceptable and opens up alternative fuel cycle pathways" with different reactor choices and use of lower enriched uranium (<20 percent) rather than highly enriched U or Pu, eliminating the need to reprocess LWR spent nuclear fuel for closed fuel cycle startup.
- "The most important fact to keep in mind in considering any estimate of the cost of alternative fuel cycles is the high degree of uncertainty about key components of each cycle. First, there is uncertainty about the cost of disposing of the high level wastes from each cycle. Second, there is great uncertainty about the cost of reprocessing spent fuel and the cost of fabricating the recycled fuel. Third, there is enormous uncertainty about the construction and operating costs for fast reactors, which are at the core of many alternative fuel cycles."

Furthermore, the MIT study cautions: "A second elusive factor that can play a large role in the economic calculations is the cost of capital (discount rate)" (MIT, 2011).

A similar study by the NEA-OECD was undertaken to "assess the available knowledge from different countries on the costs of the various options for the long-term management of spent nuclear fuel and, to the extent possible, compare the cost estimates of different countries on a common basis" (NEA-OECD, 2013). Because of major differences between countries (e.g., discount rates and government subsidies), "a direct cross-country comparison of SNF/HLW management costs was not deemed feasible," but instead simulations of a generic, idealized system operating on the same three fuel cycles were carried out with specific cost input data provided by member countries (NEA-OECD, 2013). Many of the same conclusions as from the MIT study resulted from the NEA-OECD (2013) modeling:

- Total fuel cycle costs were lower for the once-through fuel cycle, increasing successively through the monorecycle and closed fuel cycles with the difference being small compared with the LCOE.
- "Cost estimates for future facilities, including repositories, entail many uncertainties, which will only be reduced as experience is gained in implementing the necessary infrastructure." Factors that dominate the relative cost estimates are uranium price, reprocessing cost, and fast reactor capital cost premium. "Overall, the uncertainties related to the full recycling option remain the largest since only sparse data are available for these systems and no commercial system is in current operation."

Notably, relatively recent studies looked at the cost of current generation reprocessing using open data for the Thermal Oxide Reprocessing Plant (THORP) facility (Recktenwald and Deinert, 2012; Schneider et al., 2009). Recktenwald and Deinert (2012) concluded, "The analysis suggests a total life-cycle cost of 2.11 ± 0.26 mills/kWh, with a 90% and 99% confidence that the overall cost would remain below 2.45 and 2.75 mills/kWh respectively. The most significant effects on cost come from the efficiency of the reactor fleet and the growth rate of nuclear power. The analysis shows that discounting results in life-cycle costs decreasing as recycling is delayed. However, the costs to store spent fuel closely counter the effect of discounting when an intergenerational discount rate is used." This result is important because it shows that the 1 mill/kWh nuclear waste fee that had been levied in the United States would be insufficient to cover even the costs of conventional reprocessing. Advanced aqueous separations that partition the higher actinides typically operate on the PUREX raffinate. These would be additions or adaptations to the conventional PUREX process and would likely only increase reprocessing costs.

The committee notes that the 2014 DOE report on the Nuclear Fuel Cycle Evaluation and Screening Study provides a cost range based on that study's simulations:

> Alternatives to the current U.S. fuel cycle in the promising Evaluation Groups require R&D to bring the enabling technologies up to the level of successful engineering demonstration including pilot-scale facilities, which the Study results indicate as requiring several billion dollars over 10-25 years. Similarly, further development up to the first-of-a kind commercial facilities would require an additional several billion dollars. Any transition to a new fuel cycle would take decades to achieve, although some fuel cycle performance benefits such as wastes destined for deep

> geologic disposal would accrue more quickly. Fully deploying an alternative fuel cycle would likely require several hundred billion dollars or more, comparable to the cost of continuing with the current U.S. fuel cycle as new reactors replace existing reactors. (Wigeland et al., 2014)

Importantly, the committee notes that "the promising Evaluation Groups" are in fact closed fuel cycles with multirecycling: "EG23—continuous recycling of U/Pu in fast reactors," "EG24—continuous recycling of U/TRU with fast reactors," and "EG-30—continuous recycle of U/TRU in both fast and thermal reactors"[42] (all cases are with new natural uranium fuel).

Like all such studies, the specific methodologies vary, and there are many important assumptions made regarding the input data used to calculate costs, making cost comparisons challenging. A common conclusion reached across many cost and modeling studies is that, while spent fuel management represents a relatively small fraction of the total LCOE, differences in that area could result in large absolute costs depending on the size of the nuclear program and the duration of electricity generation.[43]

Studies such as those described can go only so far using economic simulations without real-world data to validate them. In the United States, more than 40 years have elapsed since the most recent commercial reprocessing activity. Despite the lack of recent actual commercial operating experience, other recent experiences in the United States shed light on the challenges of fuel cycles' implementation. For example, Chapter 5 discusses the U.S. government's less-than-stellar experience with large construction projects—specifically, the actual versus projected cost overruns (more than tens of billions of dollars) at the Hanford Waste Treatment Plant (vitrification facilities) and the canceled mixed oxide fuel fabrication facility at the Savannah River Site. Chapter 5 also provides information on the additional costs to the government when there are delays in opening geologic waste repositories; for example, about $600 million annually is being paid out of the Judgment Fund to utilities for costs of continued storage of spent nuclear fuel at nuclear power plant sites, since they cannot be stored at a disposal facility. Unlike simulations, delays and interruptions to actual fuel cycles can produce adverse effects such as backlog of spent fuel waiting for disposal and buildup of separated plutonium (as experienced by France, Japan, and the United Kingdom) when there are not enough (or none in the case of the United Kingdom) reactors using the material (see Table 2.1 in Chapter 2 that lists the holdings by country of civil separated plutonium).

In the early stages of its study, the committee became aware that cost data were considered proprietary by the reactor developers, especially with regard to reactor designs. Because the focus of this committee's study is the supporting fuel cycles and not the reactors themselves, the problem was further exacerbated by the lack of information provided by reactor developers on fuel cycle components required to support their designs, with the exception of the need for HALEU for almost all proposed designs and facilities for fabricating HALEU-based fuels. In addition, because the primary criteria and functional requirements for specific fuel cycles to support advanced reactors are either not defined or defined at such a high level, the committee was unable to find sufficient, publicly available, third-party cost information to make reliable cost comparisons. As a result, the committee was unable to conduct a true economic analysis of specific fuel cycles to support various advanced reactor designs.

As discussed in Chapter 3, advanced reactor developers claim that small modular reactors—generating notionally less than 300 MWe with water- or non-water-cooled designs—offer many advantages due to smaller and simpler designs with shorter deployment schedules, compared with other advanced reactor designs. Among the advantages claimed are scalability; load-following flexibility; ease of remote siting; and, in particular, factory-built, modular construction with lower capital construction costs compared with those of large power reactors. If true, this potential cost advantage might make small modular reactors cost competitive relative to nonnuclear energy sources. While this type of analysis is the purview of the parallel National Academies study, the committee calls attention to this economic issue because it impacts the viability of small modular reactors for commercial deployment. Given that none of the proposed small modular reactors have yet to reach the operational demonstration

[42] The sentence was revised following a prepublication version of the report to correct quoted text from the cited source.

[43] The potentially significant capital expenditures associated with management of spent nuclear fuel, such as construction of reprocessing and storage facilities, are amortized over a long period of time based on the period of electricity generation and waste storage. This limits the impact of large absolute costs on the annualized LCOE.

phase, let alone commercial deployment, real cost data is obviously lacking. Commercial viability will depend on understanding whether there is an optimal size for a small modular reactor from an economic point of view and when the break-even point will be reached for the construction of an nth-of-a-kind reactor for a particular type of small modular reactor to become economically competitive. In other words, the learning curve for both small modular reactor construction costs and deployment needs to be understood. Of particular concern to this committee is the impact of small modular reactors on the fuel cycle with emphasis on waste management and disposal, whether once-through or closed fuel cycles are used.

To gain some insights regarding costs, the committee took a graded approach that reflected the reactor choices and fuel cycle options provided by the advanced reactor developers interviewed by the committee. Most of the nonwater-cooled advanced reactor developers expressed plans to use a once-through fuel cycle combined with direct geologic disposal for the foreseeable future with the option of transitioning to a closed fuel cycle at a later time; only a few advanced reactors expressed plans to close their fuel cycles upon initial deployment. See Table 3.1 in Chapter 3 for details for each reactor type and fuel cycle choice.

Because of its base technology similarity to commercial LWRs, the development of small modular integral pressurized water reactors (iPWRs) represents the least costly and most near-term path for developing and licensing a specific reactor technology to the point where assessing its economic viability would be meaningful. With the exception of a geologic repository, all of the fuel cycle facilities required to fuel and manage the waste via interim-to-indefinite storage for iPWRs already exist.

Demonstrating one or two promising nonwater-cooled reactors using a once-through cycle would require the addition of two new fuel cycle components—specifically, enrichment facilities capable of producing HALEU (up to 20 percent uranium-235) and fuel fabrication facilities capable of handling the higher-enrichment fuels. As with current enrichment facilities for low-enriched uranium, the new enrichment facility would likely be based on existing centrifuge technology. The associated cost of a separative work unit is expected to be greater than the costs for existing low-enrichment facilities that support the LWR fleet, in order to accommodate the higher level of enrichment; such a facility would need to conform to radiological and safety standards, including criticality required for Category II special nuclear facilities.

Currently, two Category I facilities (see Section 4.2.3.1 for more details) are licensed in the United States for greater than 20 percent–enriched material; these facilities could perform both the downblending of highly enriched uranium to produce HALEU and fabrication of HALEU-based fuels, although depending on the specific fuel type, some facility modifications may be necessary for fuel fabrication. Notably, for HALEU-based TRISO fuel, two companies, BWXT and X-energy, have already taken steps to manufacture this fuel type in the United States (see Section 4.2.4.1 for details).

Additional costs will likely result from the imposition of chemical, radiological, and criticality safety standards important for storage and transportation of HALEU-containing materials while onsite at fuel fabrication facilities. All of these changes will also be accompanied by increased costs related to a higher level of nuclear material control and accounting and physical security requirements. These requirements may make it difficult for current Category III facilities to transition to Category II without significant modifications and retrofits or substantial redesign of the existing facility. It might be most cost effective to design, construct, and license a new Category II facility for fuel fabrication, rather than to upgrade an existing Category III facility to Category II and amend its license. Additionally, critical infrastructure is needed to support R&D and fuel qualification for advanced reactors, such as material testing capabilities, as discussed in Chapter 3.

Higher levels of spending will be required to support fuel cycles for nonwater-cooled reactors using a once-through fuel cycle compared with iPWRs that also use once-through fuel cycles. This prompts the question, Who should pay for the infrastructure and fuel cycle facilities to support the development of advanced reactors to the point where their economic viability could be firmly established? Notably, when President Ronald Reagan lifted the ban on commercial reprocessing in 1981, he also stated, "It is important that the private sector take the lead in developing commercial reprocessing services" (Reagan, 1981). To date, as discussed in Chapter 2, commercial reprocessing has not been economically viable in the United States.

Advanced reactor concepts that aim to close the fuel cycle by recovering and reusing the fissile material in the fuel and/or breeding new fuel require construction and operation of fuel reprocessing and fuel fabrication

facilities, with the exception of some molten salt reactor designs. Reprocessing and recycling are highly dependent on the choice of the type of reactor technology. Reprocessing will likely be a costly addition to the fuel cycle, and notably, a single reprocessing technology will not support the wide array of advanced reactor designs. For example, developers of solid-fueled, gas-cooled fast breeder reactors propose to use different reprocessing technologies to close their fuel cycles than those of solid-fueled, liquid metal–cooled fast reactors. The former are planning on a cyclic, dry pyrochemical oxidation/reduction process for actinide recycling from uranium carbide or nitride fuels, while the latter are proposing to use a nonaqueous electrometallurgical process for actinide recycling from metallic uranium or uranium alloys.[44] See Section 4.3.6.4 for more details of these processes.

Similarly, fuel fabrication facilities for the type of reactors just described will be different and use different technologies because the fuel types are different and will depend on whether (1) a homogeneous or heterogeneous approach is chosen for recycling the minor actinides and whether (2) blanket fuel is used for breeding. In any case, these facilities will require additional funding for more shielding and remote handling to ensure worker safety.

Molten salt reactors (MSRs) have very different fuel cycle requirements from those of liquid metal– and gas-cooled fast reactors operating on a closed fuel cycle. Section 4.3.6.5 describes fuel cycle processes relevant for MSRs, and the essential points are highlighted here to illustrate what is required for understanding these advanced fuel cycles. Liquid-fueled and -cooled MSRs are unique in that the molten salt acts both as the fuel and as the heat transfer fluid; no fuel fabrication is required. They are adaptable to a wide range of fuel cycles, given that breeding can occur using both thermal and fast neutrons. Because there is no fuel cladding to contain fission products, MSRs need to have off-gas treatment capabilities to continuously manage volatile effluents. For MSRs that operate with salt or fuel processing, what amounts to a chemical process facility (CPF) is attached directly to the reactor. Performance of the reactor and the CPF are intimately and complexly coupled for MSRs. The CPF serves as a reprocessing plant for MSR fuel and manages the chemistry of the reactor's molten salt for optimal reactor performance. Some MSRs operating as breeders use a one-fluid system in which the fissile and fertile materials are in the same fluid, or a two-fluid system, which is a more complex design with separate fluids for fissile and fertile materials. Some MSRs designs have no salt processing while other designs are for breed-and-burn reactors. Because of the large number of MSR designs and supporting fuel cycle options, making cost estimates is impractical until one or two specific designs emerge and their fuel cycle requirements can be defined. The cost of fuel cycles to support advanced fast reactors or molten salt reactors will be specific to the reactor technology and will potentially exceed several tens of billions of dollars as the number of deployed reactor designs increases.

During this study, the committee recognized the important concept of trade-offs when assessing potential merits and viabilities of different advanced reactors and their fuels and fuel cycles. All of the advanced reactors first need to be built and operated for a sufficient time before their technical viability can be meaningfully assessed. As these advanced reactors move from conceptual design to engineering demonstration, the supporting fuel cycle requirements need to be defined concurrently to allow for a credible economic analysis. *The U.S. government and industry going forward will have to decide which features or attributes of advanced reactors best align with U.S. energy needs without increasing proliferation risks, having an adverse impact on the environment, or imposing an unacceptable economic burden on current and future generations.*

4.5 FUEL CYCLE SAFETY CONSIDERATIONS

The radiological safety of any nuclear fuel cycle operation or facility is ensured by (1) maintaining subcriticality; (2) appropriately containing the radionuclides involved; (3) removing decay heat and preventing autoignition (fires); and (4) radiation shielding. In addition, safety requirements for fuel cycle facilities are consistent with those of other industrial chemical facilities. This section provides a brief discussion of safety in the current once-

[44] Sodium-cooled fast reactors in the French program, such as Phénix and Superphénix, used uranium oxide fuels that were reprocessed using the well-known aqueous process, PUREX. The recovered plutonium was used to fabricate fast reactor mixed oxide fuel.

through LWR fuel cycle and then describes in greater detail safety considerations associated with the deployment of advanced fuel cycles. Particular emphasis is placed on safety considerations for the operation of fuel processing facilities. Safety aspects of advanced reactor designs will be discussed in the parallel National Academies report *Laying the Foundation for New and Advanced Nuclear Reactors in the United States*.

4.5.1 Safety of the Current U.S. Nuclear Fuel Cycle

The mining, conversion, enrichment, fuel fabrication, and transportation of uranium oxide fuels and, to a lesser extent, mixed oxide fuels have occurred safely over the past 50 years. The industry and technology have developed to improve both operational efficiency and process safety in these facilities. Effluent releases of radioactivity from front-end processes are significantly below those of operating nuclear power plants (National Research Council, 2012). The highest safety risk is the release of gaseous uranium hexafluoride due to leaks or handling incidents. Normal discharges are relatively low and represent a low risk to the public.

The current open LWR fuel cycle in the United States is characterized by the use of oxide fuels and the absence of spent nuclear fuel processing. As discussed in Chapters 2 and 5, the lack of a geologic repository requires that spent nuclear fuel be stored at reactor sites throughout the country. The spent fuels currently being stored are chemically stable and maintain the fission products and actinides in a nondispersible form. Fissile material remains in a fixed geometry in which heat removal and subcriticality can be ensured throughout a variety of conditions and postulated events. Moreover, once spent nuclear fuel is moved to dry cask storage, the safety functions (prevention of criticality, protection from external hazards, decay heat removal, and confinement) are maintained through passive mechanisms. There is a long history of safe storage of these fuels. As such, there are essentially no environmental releases, and the risk of inadvertent criticality or release of radioactive material is low. Chapter 5 discusses the safety aspects of associated with storage and transportation in greater detail.

4.5.2 Safety Considerations for Advanced Fuel Cycles

The choice of fuels and fuel cycles for advanced reactors introduces additional safety considerations. In considering the safety advantages inherent to advanced reactors and their passive safety features, it is important to recognize that other processes introduced by the associated choice of fuel cycle may pose additional risks not present in the once-through uranium fuel cycle. As discussed further below, introducing processing into the nuclear fuel cycle increases the risk of a release of radioactive material into the environment due to the physical and chemical processes used in conjunction with dispersible radioactive material resulting from extended reactor operations. Recycled uranium has significantly higher gamma activity than natural uranium because of trace fission products and the presence of the decay daughters of uranium-232 (half-life = 1.9 years), which continue to build up to equilibrium in approximately 10 years. Recycled uranium also has higher alpha activity than natural uranium from the presence of uranium-234 and traces of plutonium and neptunium. In addition to increased radiation exposure from these decays, the decay products also generate additional heat in excess of that from natural uranium. If the feedstock also contains plutonium, additional gamma activity will result from the buildup of americium-241. The source term for reprocessed fuels will, thus, contain other radioactive isotopes and decay products that must be considered in the evaluation of on-site and off-site consequences. Fuel cycle facilities must implement safety features to protect workers from these hazards, as well as designs to minimize the risk to the off-site public and environment. Several presenters to the committee discussed the safety features of their advanced reactor designs, as well as the potential waste reduction and fuel utilization benefits of closing the nuclear fuel cycle through fuel reprocessing. However, no presenters discussed advances in the safety of processing facilities, such as passive cooling or inherently safe processing techniques, suggesting that the safety considerations have been limited to operational reactors with little attention paid to improving safety margins in the rest of the fuel cycle. In fact, by introducing reprocessing into a nuclear fuel cycle, it is not clear to what extent the safety benefits of new reactor technology are offset by the increased safety risk inherent to fuel reprocessing.

4.5.2.1 Safety Considerations for the Front End of the Fuel Cycle

On the front end of the fuel cycle, additional criticality controls are required for processing HALEU to manage risks in fuel fabrication. The potential impact of advanced fuel cycles on the safety of front-end processes results primarily from differences in the source material radioactivity and heat loading and the increased enrichments of HALEU fuel, which reduce margins to inadvertent criticality. Incidents that result in solidification must be prevented or shown to remain subcritical by geometry. An example of such an inadvertent criticality event occurred in 1999 at the JCO Fuel Fabrication Plant in Japan, when operators attempted to process uranium fuel at 18.8 percent enrichment using a technique they had successfully used previously to process uranium with 6 percent enrichment. It resulted in three workers suffering from acute radiation sickness, which was fatal in two of the cases. In addition, several other facility workers and members of the public received radiation doses, more than 100 members of the public were evacuated, and hundreds of thousands of residents of the prefecture were instructed to shelter in place for 18 hours because of the off-site release of radiation (IAEA, 1999a).

Moreover, the neutron emissions of even isotopes of plutonium, particularly plutonium-238, must be accounted for to reduce potential for criticality and limit radiation doses to workers. In addition to controlling the material's amount and geometry, careful design of facilities should preclude the introduction of moderation and reflection, particularly from water ingress. As such, design and controls should minimize the risk from both internal (fire piping and other liquid sources) and external flooding.

4.5.2.2 Safety Considerations for Fuel Processing

On the back end of the fuel cycle, spent nuclear fuel processing introduces risk associated with complex industrial chemical or electrochemical processes that involve treating significant quantities of fissile material with large quantities of other hazardous material. Complex industrial chemical processes also introduce additional potential failure modes; for instance, reprocessing facilities can experience failures involving inadvertent criticality, gaseous and liquid leaks, fires, and adverse chemical reactions, as well as susceptibility to external hazards and prolonged loss-of-power events. Additionally, although processing operations are generally conducted at lower temperatures and pressures than reactor operations, they involve active handling of highly radioactive fissile material, fission products, and other material in dispersible form that is subject to physical and chemical (and often vigorous) processes, as shown in Table 4.1; the associated risks to site workers, off-site populations, and the environment must be properly managed. Improvements in technologies and processes have reduced these risks over time; for example, between 1985 and 2005, the average annual occupational radiation exposure at reprocessing facilities decreased from 10 mSv (millisievert) to 1.5 mSv per person (IAEA, 2005c).

Potential safety hazards associated with fuel reprocessing facilities include release of radioactive materials and accumulation of pyrophoric or highly reactive species. For example, mechanical processes at the head-end release nonretained gases, such as radioactive krypton-85 and iodine-129, and can accumulate pyrophoric metals (such as zirconium) from fuel shearing and cladding removal. Additional processes can release radioactive tritium, technetium-99, several iodine isotopes, carbon-14, and nitrogen-15. The separation of fissile material in many processes involves dissolution in nitric acid and chemical treatment of pyrophoric material, reactive chemicals, and flammable solvents. In addition to increased radiation exposure from the decay of isotopes (such as uranium-232, uranium-234, americium-241) inherent in reprocessed fuel, the decay products also generate heat in excess of that from natural uranium.

Throughout all of the processes, decay heat and other sources of self-heating must be effectively removed to ensure the chemical and physical form of various process streams are maintained within controllable and predictable bounds. The presence of fissile material requires deliberate management of material inventories and geometries in order to ensure that adequate margin to subcriticality is maintained. The accumulation of undesirable species in recycled reagents and high-level waste must also be carefully managed to avoid severe overheating. For example, in processes such as PUREX using concentrated nitric acid or heavy metal nitrates (uranyl or plutonium nitrate), an organic (such as tributyl phosphate) and kerosene can combine in a radiation flux to produce nitrated organics often referred to as red oil. If process and chemical conditions are not properly maintained, violent nitration oxidation reactions involving red oil can occur. Such red oil events have occurred (Savannah River Site in 1953

and 1975 and Tomsk, Siberia, in 1993) when organic solutions and kerosene used in PUREX processing mix with nitric acid at high temperature (>120°C). Modern fuel cycle facilities have effectively minimized the explosive risks from red oil, as well as hydrogen (from radiolysis) and other explosive gases resulting from operations, by controlling temperature and other process parameters within safe bounds and limiting buildup of susceptible material through control of the reaction rate and use of engineered controls.

Often, process and facility design require balancing competing safety considerations. For example, liquid mixing of fissile material reduces hazards associated with radiotoxic dust, but it requires additional attention in the design to prevent inadvertent criticality. Thorium fuel reprocessing is complicated by the presence of crystalline ThO_2 and unirradiated PuO_2, both of which are difficult to dissolve in nitric acid. Fluorine can be added to improve dissolution, but doing so can increase the possibility of leaks and vessel failures because of its corrosive nature and incompatibility with common materials in reprocessing equipment (IAEA, 2019e).

Because of the risks associated with processing spent nuclear fuel, a graded approach to defense in depth[45] is applied to processing facilities, similar to that applied to the nuclear power plants they support (IAEA, 2017b). Processing facilities are properly sited to minimize risks of external hazards (fires, floods, and seismic events) and limit the off-site population and areas affected by potential radioactive releases. On-site risk is minimized through careful process and facility design and the use of codes and standards to ensure high quality of construction. Such design includes heavy radiation shielding and passive geometries to prevent inadvertent criticality. Facility design must include passive features, automated controls, and operators that maintain parameters within specific bounds during normal operations and in response to anticipated operational occurrences. For example, inert, leaktight enclosures are used to minimize explosive risks and protect personnel from radiotoxicity associated with plutonium dust. Facility designs must include liquid and gaseous radioactive waste processing for controlling radioactive effluent releases incidental to operations within regulatory limits. In the event of an accident, additional engineered safety systems mitigate the effects, including multilevel confinement systems designed to prevent large off-site releases of radioactive material. Some of these safety systems, such as dynamic confinements and tank cooling systems, are active systems and rely on emergency power and other support systems to function. These systems are needed to ensure that chemical processes are safely shut down, nuclear material remains subcritical, hazardous concentrations of explosive or flammable gases do not accumulate, and radiological material is cooled and confined. Should an off-site release occur, the final layer of defense in depth ensures that off-site emergency response organizations are trained, equipped, and prepared to mitigate these consequences and protect public health and the environment.

Several large processing facilities have been safely operated throughout the world, but the risk associated with these facilities is real, and accidents have occurred[46] (Bixler et al., 2017; IAEA, 1996).

- Mayak Production Association, Russia (1957)—high-level waste tank cooling system failure led to a nitration oxidation reaction and an explosion (equivalent to 74 tons of TNT) and subsequent release of over 740 PBq (petabecquerel) of radioactivity.[47] This event was classified as a Level 6, Serious Accident, the second most severe level on the International Nuclear Event Scale (INES).
- Windscale Reprocessing Plant, United Kingdom (1973)—explosion and off-site release due to an exothermic reaction in a reprocessing tank. Classified as INES Level 4, Accident without Significant Offsite Risk.

Safe operation of reprocessing facilities in the United States (within the nuclear weapons complex for material production and West Valley Demonstration Project) and internationally has demonstrated that these risks can be managed safely (IAEA, 2005c). In order to fully realize the safety benefits associated with the deployment of

[45] As defined by the U.S. NRC, *defense in depth* is "An approach to designing and operating nuclear facilities that prevents and mitigates accidents that release radiation or hazardous materials. The key is creating multiple independent and redundant layers of defense to compensate for potential human and mechanical failures so that no single layer, no matter how robust, is exclusively relied upon. Defense in depth includes the use of access controls, physical barriers, redundant and diverse key safety functions, and emergency response measures" (U.S. NRC, 2021c).

[46] For summaries and data of previous safety incidents at fuel cycle facilities, see IAEA (1996) and Bixler et al. (2017).

[47] By comparison, the accident at Chernobyl was estimated to have released approximately 1,000 to 2,000 PBq (IAEA, 1986).

existing or advanced reactors, the design of the fuel cycle processes should seek to optimize safety over the entire fuel cycle, leverage international experience, and improve safety margins through implementation of passive and inherently safe facility and process designs.

4.5.2.3 Safety Considerations for Transportation and Storage of Radioactive Materials

While the type and design of facilities vary, the processing of spent nuclear fuel includes the risks associated with transport and storage of spent nuclear fuel awaiting processing and introduces additional safety considerations beyond those of an open fuel cycle. For example, while reprocessing reduces the inventory of radioactive and fissile material ultimately disposed in a geologic repository, it requires managing large inventories of these materials at processing facilities, as illustrated in Table 4.4. In addition to the risks associated with the processing of spent nuclear fuel, processing facilities will also serve as interim storage locations for spent nuclear fuel awaiting processing and for the storage of low- and high-level radioactive wastes resulting from reprocessing. The risks to spent nuclear fuel would be the same as when stored at individual reactor sites, but would occur in larger quantities, potentially putting more material at risk to external events.

Several vendors noted that they would be using sodium-bonded metallic fuel for initial demonstration or until they develop more advanced metallic fuel designs. Sodium is highly reactive and pyrophoric in both air and water even at relatively low temperatures. These reactions are exothermic and generate caustic sodium hydroxide and explosive H_2, which presents challenges to qualification for storage both interim and in a deep geologic repository. Therefore, in the absence of innovative storage or disposal options not currently available, processing is required, even for a once-through fuel cycle, to avoid adverse chemical reactions or explosions, which—in combination with dispersible radioactive fission products, actinides, and other material—would result in significant environmental contamination. Special procedures and safety protocols must be implemented in processing sodium-bonded fuels and other sodium-wetted components. Sodium must be removed before working on materials, as even residual sodium will react with air and cause fires that generate caustic aerosols. Nitride and carbide fuels are also pyrophoric and require similar safety protocols (IAEA, 2007b).

4.5.2.4 Operational Experience: Effluents

The separation of fissile material during spent nuclear fuel processing also generates other waste streams. Some of these waste streams are captured in waste forms for disposal in low-level waste disposal facilities or a geologic repository, while others are released to the environment either in liquid or gaseous forms. Of particular concern for processing facilities is the release of ^{3}H, ^{85}Kr, ^{129}I, ^{14}C, ^{15}N, and traces of alpha emitters including Pu.

^{85}Kr, a noble gas with a 10.76-year half-life, represents the most significant release of gaseous radioactivity incidental to normal operations. Kr is released during head-end operations and throughout the separation stages. While bench-level methods for removing ^{85}Kr have been explored, none have been demonstrated at an industrial level (Croff et al., 2008). Krypton is not, therefore, removed by off-gas treatment in industrial facilities. Because the quantities released exceed the practical capacity of current removal and retention technologies, retention is not practical, and ^{85}Kr is released directly to the atmosphere (NEA-OECD, 2005).

Iodine, with ^{129}I as the predominant isotope of concern, is scrubbed in off-gas treatment systems that involve passing it through a sodium hydroxide solution or capturing it in charcoal adsorber beds and filter trains. Internationally, iodine captured in sodium hydroxide has usually been discharged as liquid radwaste to the ocean. Design of reprocessing facilities in the United States has captured iodine in other media and retained for disposal.

Tritium (^{3}H) is released to the environment, either directly via the liquid waste pathway or evaporated and released via the ventilation pathway.

In the United States, 40 CFR 190 provides federal limits for the total quantity of "radioactive material entering the general environment from the entire uranium fuel cycle." Limits are provided per gigawatt-year of electrical generation for ^{85}Kr, ^{129}I, and ^{239}Pu, and other alpha emitters. Table 4.4 below compares discharges from the La Hague facility in 2008 to calculated 40 CFR 190 limits and typical releases from U.S. nuclear power plants. The

most significant release from the La Hague facility is ^{85}Kr, and such levels would challenge current U.S. regulatory limits should a similar facility be built and operated in the United States. Tritium releases from processing are comparable to those from a single nuclear power plant.

TABLE 4.4 Comparison of Radioactive Material Release Limits to Typical Discharges at the La Hague Reprocessing Facility and a Typical U.S. Nuclear Power Plant

	40 CFR 190.10 Limit (per GWe-yr)	40 CFR 190.10 Limit at 45 GWe-yr[a,b]	La Hague effluent releases (2008)[a]	Typical Nuclear Power Plant Effluent Release per Year[c]
^{85}Kr	1.85×10^6 GBq (50,000 Ci)	8.30×10^7 GBq	1.55×10^8 GBq	Total Noble Gas 1×10^1 GBq to 1×10^5 GBq
^{129}I	0.185 GBq (5 mCi)	8.32 GBq	6.76 GBq	
^{239}Pu and other alpha emitters	0.018 GBq (0.5 mCi)	0.81 Gbq	1.83×10^{-3} GBq	
Tritium			4.64×10^4 GBq	BWR: 1.1×10^3 GBq PWR: 1.85×10^4 GBq

[a] Data from Van der Stricht and Janssens (2010).
[b] Based on a total of 63.2 GWe of nuclear capacity in France in 2008 running at a 72 percent capacity factor.
[c] Data from National Research Council (2012).

5

Management and Disposal of Nuclear Waste from Advanced Reactors

This chapter responds to the charges in the statement of task that call for evaluating the waste management and disposal options for the various proposed advanced nuclear reactors. The committee conducted this evaluation by accounting for typical volumes and physical, chemical, and isotopic characteristics of waste streams, including from possible reprocessing, from these advanced nuclear reactor technologies, and examining transportation, storage, and ultimate disposal requirements for these wastes.

The committee first provides the summary, findings, and recommendations (Section 5.1), and then describes the U.S. nuclear waste management and disposal program (Section 5.2), discusses the concept of geologic disposal of highly radioactive nuclear waste (Section 5.3), describes what the committee learned from experts' briefings (Section 5.4), delves into the specific waste issues that arise from advanced nuclear reactors and fuel cycles (Section 5.5), assesses the potential impacts of advanced nuclear fuel cycle wastes on storage and transportation operations (Section 5.6), and provides an overview of decontamination and decommissioning of nuclear power plants relevant for this study (Section 5.7). Throughout the chapter, the committee provides observations on waste management and disposal for advanced nuclear reactors and fuel cycles.

5.1 CHAPTER 5 SUMMARY, FINDINGS, AND RECOMMENDATIONS

As described in Chapters 3 and 4, the development of advanced nuclear reactors would require significant efforts to establish a supporting fuel cycle. Broadly, there are two fuel cycles: open and closed. A closed fuel cycle involves reprocessing and reuse of fissile material, mainly uranium and plutonium, in order to harvest some of the remaining energy content of spent fuel. Chapter 4 deals extensively with different strategies for closed fuel cycles. In contrast, an open fuel cycle is based on the direct, permanent disposal of spent fuel after irradiation, as well as the high-level waste that may result from chemical reprocessing. Hence, for an open fuel cycle (for the purposes of this report), spent fuel is waste.[1] In the United States, the present strategy is an open fuel cycle with direct disposal of spent fuel in a geologic repository. Indeed, the proposals for advanced reactors are based on an open fuel cycle and disposal of the many different types of fuels that are now proposed or considered. As concluded in the 1996 National Research Council report *Nuclear Wastes: Technologies for Separations and Transmutation*, regardless of whether a country adopts an open or closed fuel cycle, a geologic repository is required.

[1] Here, a monorecycling fuel cycle that results in spent mixed oxide fuel that will be disposed of is considered an open cycle.

Section 5.2 sets the stage with a historical review of U.S. nuclear waste management, highlighting the poor planning, continued programmatic delays, and failure to create a geologic repository. It then outlines key concepts, strategies, and challenges for geologic disposal to give readers an understanding of the relevant factors and problems that advanced reactors should target if their goal is to mitigate nuclear waste management and disposal challenges. Advanced reactors that change the inventory of actinide isotopes by using new fuel types, increased levels of enrichment, higher burnup, or actinides from reprocessing will not be solving a problem that could not be addressed by selection of a repository site with appropriate geochemical conditions and geologic environment and may only contribute to waste disposal problems by introducing new, complex waste streams that require application of additional technology before disposal.

The subsequent sections highlight these unique waste considerations for different reactor types during all stages of the fuel cycle and for all levels of waste. For advanced reactors, the different types of fuel, burnups, enrichment levels, and processing steps for the fuels being proposed raise new questions about waste management and disposal requirements that are not well understood or well studied by reactor developers. Per the committee's statement of task, this chapter evaluates the various management and disposal options for the fuel from advanced reactors; accounts for typical volumes and characteristics of waste streams from advanced reactors; and examines the transportation, storage, and disposal requirements for these wastes. Key issues include managing large volumes of irradiated graphite, handling and developing stable waste forms for sodium-laden and molten salt waste streams, and developing storage and transport packaging for fresh and irradiated high-assay low-enriched uranium (HALEU) fuels. Overall, the implementation of advanced reactors and fuel cycles will not solve problems associated with nuclear waste disposal, and may even exacerbate them depending on the nature and quantity of the waste streams generated.

From these analyses, the committee made the following findings and recommendations:

Finding 11: As the United States nears the 40th anniversary of the Nuclear Waste Policy Act (NWPA) (Public Law 97-425) and its Amendments (Public Law 100-203, Part E), there is no clear path forward for the siting, licensing, and construction of a geologic repository for the disposal of highly radioactive waste (mainly commercial spent nuclear fuel). The United States finds itself in this difficult situation for many reasons, including (1) changes to the original NWPA of 1982 that moved the process of site selection from a consideration of multiple sites to a single site, Yucca Mountain, Nevada; (2) a slowly developing and changing regulatory framework that provided late guidance in the site selection process and the evaluation and comparison of multiple sites; (3) ineffective management of the Nuclear Waste Fund ($45 billion) by Congress, which treated what was to have been a ratepayer escrow account as if it were taxpayer monies; (4) consequential policy changes occurring with changing administrations; (5) conflicting congressional and executive policies; and (6) insufficient public engagement in decisions concerning the basic strategy for the storage and disposal of the waste. The continued delay in planning and progress has only made the situation more complicated, as the present legal and regulatory frameworks have become outdated and even more limiting. Numerous assessments during the past decade, notably the *Blue Ribbon Commission on America's Nuclear Future* (2012) and *Reset of America's Nuclear Waste Management Strategy and Policy* (2018), have outlined a way forward. The committee agrees with common recommendations of these studies to establish a single-mission nuclear waste management and disposal entity, for which models have been proposed that deserve consideration by Congress. The entity could be governmental, partially governmental, or private; as to the latter option, the committee notes that two successful programs are being led by fully private entities: Posiva in Finland and SKB in Sweden. Important attributes of the entity are described in Recommendation G.

Recommendation G: Congress should establish a single-mission entity with responsibility for the management and disposal of nuclear wastes.

- **Such an entity should be responsible for "cradle-to-grave" care and disposition of spent nuclear fuel—that is, from its discharge from a reactor plant to its final disposal in a repository. This entity should have continuity of leadership and funding, as well as a consistent disposal strategy. It should also have high technical and scientific competence, be able to organize and lead research programs and large**

construction projects, and, importantly, be able to engage the public in a way that engenders trust. Finally, the entity should operate effectively over the many decades that will be required to manage the present inventory of nuclear waste, as well as waste generated by future advanced reactors.

- **Congress should ensure that funds collected from ratepayers that use electricity from nuclear power plants, now over $45 billion, are applied to the disposal of the spent fuel generated by nuclear power plants and that collection of funds from all commercial generators of nuclear power resumes. Moreover, funding for the entity should be held in a true escrow account and not be subject to the annual appropriations process.**
- **The entity should immediately initiate steps to begin the process of site selection. Before sites are considered, a decision-making process with appropriate technical criteria and an acceptable method of public engagement, such as consent-based siting, needs to be defined in collaboration with impacted communities, tribes, and states. Congress should make a decision on what to do with Yucca Mountain, which could include keeping it as a possible site for consideration, depending on the plans of the new entity.**

Finding 12: The advanced reactor developers' presentations to the committee focused on the reactors themselves, with little or no attention to nuclear waste management or disposal of the nuclear waste generated because there is no incentive for them to do so. In the absence of a final geologic disposal strategy in the United States, the expansion of nuclear power using advanced reactors will add to the amount of spent nuclear fuel and associated waste that requires disposal and increase the complexity of this challenge because of the need to dispose of new types of fuels and waste streams.

Finding 13: Presently proposed advanced reactor technologies will initially use a once-through fuel cycle; however, compared with those currently in use, the fuels will have a higher uranium enrichment (e.g., high-assay low-enriched uranium [HALEU]) and a higher burnup; also, they will use new types of fuel materials and designs (e.g., TRistructural ISOtropic [TRISO] fuels). As compared with the disposal of the present uranium oxide spent fuel, these new fuel types may result in changes of (1) the amounts (either in mass or volume), chemical compositions, and radionuclide inventories of the waste to be disposed; (2) the thermal power of fuel assemblies; and (3) the durability of the spent fuel in a disposal environment. More specifically, from the waste management and disposal perspective, it is important to note the following:

- Radiological risks from disposed waste are dominated by the mobility of long-lived radionuclides and not by the radiotoxicity inventory. Therefore, radiotoxicity itself is a poor metric for repository performance and risk to the public from waste disposal. The long-term safety of disposal of actinides in appropriate geologic settings is largely independent of the actinide inventory of the repository, except in the off-normal situation where the geologic barrier is bypassed—for instance, by human intrusion. Because the amount of mobile long-lived fission products generated is independent of reactor type, most advanced reactor technologies will have little impact on estimates of long-term repository performance. Key factors for long-term repository performance are the redox conditions of the geochemical environment, waste form stability, groundwater flow rates, and solubility/sorption of radionuclides. A reducing environment is preferred. Advanced reactor technologies will have little or no impact on these factors.
- The total quantities of fission products generated are generally related to fission rate and are largely independent of reactor technologies, although the distributions of different isotopes may differ. Both short- and long-lived fission products are important on the timescales relevant to geologic disposal. Short-lived fission products (e.g., strontium-90 and cesium-137) produce significant heat, while long-lived fission products (e.g., iodine-129 and technetium-99) are extremely mobile in a repository environment. Advanced reactor technologies will, in general, generate a higher amount of fission products in each spent nuclear fuel package because of their higher burnups, resulting in a higher thermal load. Increased thermal loads of waste containers will impact a number of repository design features, such as the size and spacing of waste packages, the size of the repository footprint, and engineering designs, thereby impacting the cost of repository construction.

- Enhanced stability and durability of waste forms in a repository environment can be beneficial to the performance of a repository by limiting the release of radionuclides from the spent fuel. Some advanced reactor technologies propose using advanced fuel designs with the potential to contain radionuclides (e.g., TRISO fuel), but this potential must first be demonstrated by experimental programs that examine the fuel's long-term integrity in intense radiation fields and at high temperatures.

Recommendation H: The implementer of the nuclear waste management and disposal program, in collaboration with advanced reactor developers, should support research and development on (1) spent fuels from advanced reactors to understand their degradation behaviors in a variety of geologic environments, (2) recycling and reuse options for irradiated graphite, and (3) management and disposal of unique waste streams from advanced reactors that may pose a challenge for geologic disposal. Moreover, the wastes and treatment technologies should be characterized and quantified.

Recommendation I: The principal agencies (U.S. Department of Energy, U.S. Nuclear Regulatory Commission, and U.S. Environmental Protection Agency) should initiate a coordinated effort to develop regulations and standards for a generic repository (i.e., not specific to Yucca Mountain) and new types of spent fuel and waste forms in order to support geologic disposal of new fuel types from advanced reactors. Developers of advanced nuclear reactors also need to anticipate the impact of new fuel types on their performance as a waste form in a geologic repository.

Finding 14: Conceptually, advanced reactors could be used to reduce the current inventory of transuranics in the approximately 86,000 tonnes of legacy spent fuel to date; this would require considerable resources and time to design, develop, prototype, build, and make operational the required infrastructure. Creating this infrastructure is not practicable in the near future, as long as uranium and enrichment services are readily available.

Recommendation J: The immediate-future focus of the U.S. nuclear waste management and disposal program should be planning for the geologic disposal of the existing spent fuel that is presently stored at 79 sites in 35 states and the approximately 2,000 metric tons per year being generated by existing commercial reactors.

Finding 15: Most of the advanced reactor types proposed would generate waste streams for which there is little experience or mature technical ability to manage. All additional waste treatment options would entail additional costs not encountered in the management and disposal of spent light water reactor (LWR) fuel. High-temperature gas reactors will produce much larger volumes of spent fuel compared with equivalent energy production from LWRs. It may be possible to reduce the volume by removing graphite from the spent fuel, but those technologies are immature. Dust production from pebble-bed reactors would pose waste and decommissioning challenges. Sodium-cooled fast reactors would produce large volumes of irradiated sodium waste that would require treatment and disposal; sodium-bonded spent fuel is not suitable for direct disposal and would require treatment by methods not yet technically mature at the industrial scale. Molten salt reactors produce two waste streams, radioactive off-gases and the spent fuel salt waste, which would require processing into waste forms suitable for disposal. These treatment methods and suitable wastes forms are in early stages of exploration. Most of these advanced reactors would produce large quantities of irradiated graphite waste—from use as moderators or reflectors—and this material would prove challenging to manage as well. While European researchers have analyzed graphite waste disposal extensively, researchers in the United States generally lack this expertise.

Finding 16: Similar to issues with waste management, advanced reactor developers have not adequately examined the back-end operational management (i.e., storage and transportation) of advanced nuclear spent fuel. Consequently, the stability of waste forms and potential issues related to needed processing prior to storage, as well as repackaging that may be required for transportation and final disposal, have not been studied sufficiently.

Finding 17: Secondary waste streams—such as lead, sodium, molten salts, and irradiated graphite (moderators and/or from TRistructural ISOtropic [TRISO]–particle fuel disassembly)—from advanced reactor and fuel cycle operations will need to be stabilized and packaged for storage prior to downstream operations to support disposal. Waste forms for these secondary wastes can be developed to be compatible with storage regulations by the U.S. Nuclear Regulatory Commission; however, some still require research and development to properly characterize performance envelopes.

Recommendation K: The U.S. Department of Energy (DOE) should require and fund advanced reactor developers to work with designers of storage and transportation concepts to mitigate potential fuel cycle disconnects caused by suboptimized designs that satisfy only one operational aspect of the back end of the fuel cycle (e.g., storage, transportation, or disposal). Through venues such as the Extended Storage Collaboration Program of the Electric Power Research Institute, DOE should continue to collaborate with industry to identify and address long-term storage packaging issues and how they may potentially impact downstream transportation and disposal operations. This recommendation applies to all wastes generated from reactor operations and potential reprocessing operations. When appropriate, DOE should consider funding research and development to address common waste form degradation issues and their impact on storage and transportation system designs. The implementer of the nuclear waste management and disposal program should execute this recommendation.

Finding 18: Because of the higher enrichments of fresh high-assay low-enriched uranium (HALEU) and potential higher burnups of irradiated HALEU fuels, maintaining subcriticality margins and having adequate thermal and shielding protection during transport and storage would most likely require at least some of the following:

- criticality experiments for enrichments above 5 percent to support benchmarking analyses;
- assessment of the feasibility of using type 30B containers for transport of enriched uranium hexafluoride, if needed; and
- criticality, thermal, and shielding assessments for storage and transportation.

Recommendation L: In its advanced reactor programs, the U.S. Department of Energy should support funding and provide technical resources for integration of high-assay low-enriched uranium (HALEU) products into advanced reactor fuel cycles by performing criticality, thermal, and shielding assessments of storage and transportation systems to meet stated schedules of deployment for demonstration and prototyping of advanced reactors.

5.2 THE U.S. NUCLEAR WASTE MANAGEMENT AND DISPOSAL PROGRAM

Though the United States has reached the 40th anniversary of the Nuclear Waste Policy Act (NWPA) of 1982, there remains no clear path forward for the siting, licensing, and construction of a geologic repository for the disposal of highly radioactive waste, mainly the spent nuclear fuel from commercial nuclear power plants and the high-level waste generated by chemical reprocessing associated with defense programs. Instead, as of 2021, the United States has accumulated about 86,000 MTHM (metric tons of heavy metal) of spent nuclear fuel generated by commercial nuclear power plants and hundreds of thousands of cubic meters of liquid/sludge waste at the Hanford and Savannah River sites where irradiated fuel was reprocessed to obtain plutonium for nuclear weapons (GAO, 2021a,b). The inventory of spent fuel from the 93 operating commercial reactors in the United States continues to increase at the steady rate of approximately 2,000 MTHM/year.

The distribution of both commercial and defense waste is shown in Figure 5.1(a). On the defense side, most of the wastes are located at facilities originally associated with the Manhattan Project and the Cold War production of fissile material for nuclear weapons. These facilities and laboratories are noted on the map, but many contaminated

areas related to defense activities are not included, as they generally have lower-activity waste, such as at the Mound and Fernald plants in Ohio, the Portsmouth uranium enrichment plant in Ohio, and the Paducah uranium enrichment plant in Kentucky. Figure 5.1(a) does show the Hanford and Savannah River sites, as well as Idaho National Laboratory, which house hundreds of large underground tanks of radioactive waste from reprocessing of spent fuel. Despite the extent of nuclear contamination related to defense sites (DOE-EM, 1997), the most important problem in terms of total radioactivity is the 86,000 MTHM of commercially generated spent nuclear fuel stored on the sites of nuclear power plants at 75 locations in 33 states, as depicted in Figure 5.1(b) (GAO, 2021a). A 2013 analysis concluded that all light water reactor (LWR) spent nuclear fuel generated to date can be disposed of, as it would not be needed as a source of plutonium for potential future fast reactors (Worrall, 2013). Clearly, the greatest source of radioactivity is in the spent nuclear fuel from commercial nuclear power plants (see Table 5.1).

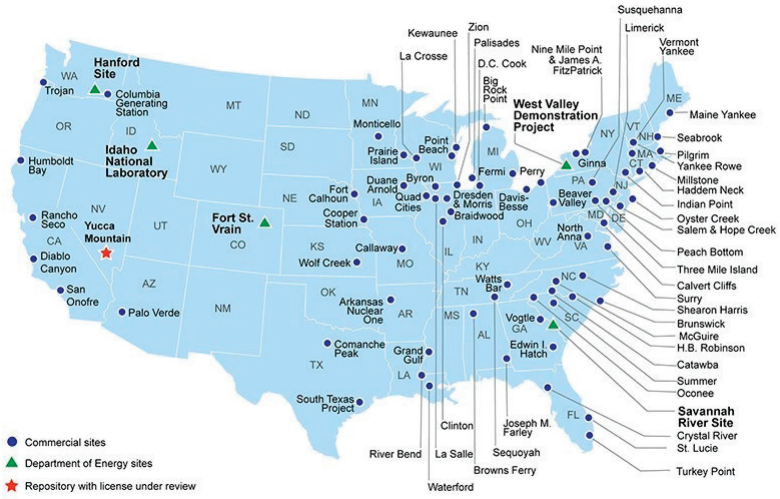
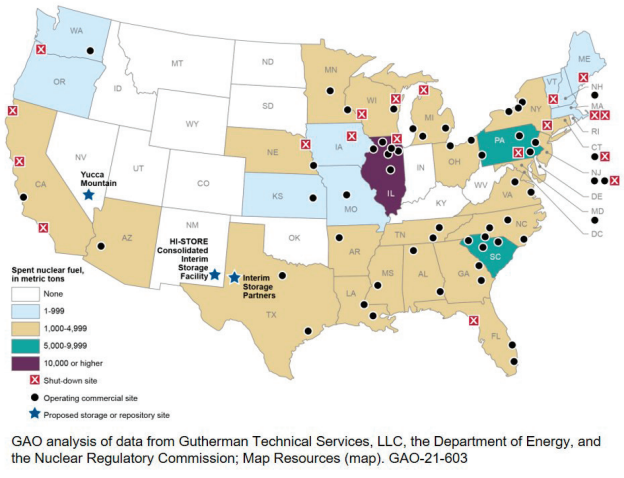


FIGURE 5.1(a) Locations of commercial and defense nuclear waste in the United States.
SOURCE: Adapted from GAO (2017).



GAO analysis of data from Gutherman Technical Services, LLC, the Department of Energy, and the Nuclear Regulatory Commission; Map Resources (map). GAO-21-603

FIGURE 5.1(b) Stored commercial spent nuclear fuel sites, showing ranges in amounts of spent fuel in metric tons for each state.
NOTE: Note that Fort St. Vrain's site is considered U.S. Department of Energy owned.
SOURCE: Adapted from GAO (2021a).

For high-level nuclear waste generated by weapons programs, the plan remains to vitrify some of the high-level waste prior to geologic disposal. Throughout the 23-year history of the Savannah River Site, some 4,100 containers of vitrified waste have been generated at the Defense Waste Processing Facility; the "vitrified logs" remain on site, as there is no geologic repository to receive them (SRS, 2020). At the Hanford site, the vitrification plant remains under construction as the schedule slips by decades and the cost escalates by tens of billions of dollars (GAO, 2020), as further discussed in Section 5.2.3. At both the Savannah River and Hanford sites, the most recent strategy involves chemical processing of the highly radioactive waste into low and highly radioactive waste streams. The low-level activity waste streams are being or will be disposed of in near-surface disposal sites. The inventory of U.S. nuclear wastes is summarized in Table 5.1.

5.2.1 Waste Classification

Any advanced reactor or advanced nuclear fuel cycle will generate nuclear waste at each step of the cycle: uranium mining and milling, uranium enrichment, fuel fabrication, reactor operation, chemical processing, reprocessing and recycled fuel fabrication, and finally decontamination and decommissioning. In the United States, highly radioactive wastes, such as spent nuclear fuel and high-level waste, are classified based on their origin. Lower-activity wastes also are indirectly source-based, as they are defined by excluding the source-based classes of fuel (BRC, 2012). At times, this chapter uses the international category of "intermediate-level waste" (ILW), which is closest to the U.S. category of Greater-than-Class C (GTCC) waste, because the available literature used such terminology. Appendix D provides more information on U.S. waste classifications, including a discussion of ILW and GTCC.

The classification of the waste determines the type of disposal environment. Spent nuclear fuel, solidified high-level waste, and transuranic waste from defense programs must be disposed of in a geologic repository. Lower-activity waste can generally be stored or disposed of in near-surface facilities. The U.S. Nuclear Regulatory Commission (U.S. NRC) has been evaluating disposal options for GTCC waste, which contains higher concentrations of radioactivity than Class C waste and may require disposal in a deep geologic repository (DOE, 2017). In 2020, the U.S. NRC released an options memo for the commissioners that supported the conclusion that most GTCC waste streams could be suitable for near-surface disposal, pending site-specific analyses, though no rulemaking has been made as of yet (U.S. NRC, 2020h).[2] More information on disposal of different waste classes can be found in Appendix D. *In order to estimate the cost, determine environmental impact, and complete a safety assessment of any advanced fuel cycle, it is necessary to know the type, amounts, and heat production of waste that will be generated.*

TABLE 5.1 Nuclear Waste in the United States as of 2020

Waste Category	Waste Inventory[a]
Uranium mine and mill tailings	438 million m^3 1.11×10^8 TBq (3,000 MCi)
Depleted uranium (UF_6)	760,000 MT[b]
High-level waste (defense reprocessing)	380,000 m^3 8.88×10^7 TBq (2,400 MCi)
Buried waste (TRU, LLW, hazardous)	6.2 million m^3
Spent nuclear fuel (commercial)	~88,000 MTHM ~2.22×10^9 TBq (~60,000 MCi)
Contaminated soil	79 million m^3
Contaminated water	1,800–4,700 million m^3
Estimated cleanup and disposal cost = $300 billion	

[a] Radioactivity values provided in Ci as well as the international standard Bq, 1 Ci = 3.7×10^{10} Bq.
[b] From DOE (2016).
NOTE: LLW = low-level waste; MCi = megacurie; MT = metric ton; TBq = terabecquerel; TRU = transuranic elements.
SOURCE: DOE-EM (1997), with updated calculation of total MCi for spent nuclear fuel, unless otherwise noted.

[2] This sentence was altered after release of a prepublication version of the report to align the text with information from the associated reference.

5.2.2 Interim to Indefinite Storage

In the absence of a geologic repository, the de facto U.S. strategy for dealing with commercially generated spent nuclear fuel is the possibility of consolidated, interim to indefinite storage. This strategy accommodates the present situation, but it is not a strategy by design. In fact, legislation will be required in order to eliminate the limitations imposed by the Nuclear Waste Policy Act of 1982, as amended in 1987, that prevents the interim storage of spent nuclear fuel prior to the opening of a geologic repository, which at that time had been designated to be at Yucca Mountain (DOE-RW, 2008).

Additionally, in 2014, the U.S. NRC replaced its revised "waste confidence rule" of 2010 with a "continued storage" rule for spent nuclear fuel. The waste confidence rule had previously been used as a statement that the U.S. NRC has "reasonable assurance that sufficient mined geologic repository capacity will be available to dispose of the commercial high level radioactive waste and spent fuel generated in any reactor when necessary" (U.S. NRC, 2010). However, in 2012, the U.S. NRC had to institute a moratorium on the issuance of licenses for new reactors and renewals of existing licenses because a federal court in the District of Columbia Circuit determined that, in the absence of progress in developing a national geologic repository, there was little basis for confidence in the federal government's ability to permanently dispose of spent nuclear fuel. The U.S. NRC then developed a new continued storage rule concluding that spent nuclear fuel could be stored indefinitely, as long as institutional controls that ensure the safety of the waste remain in place indefinitely (U.S. NRC, 2014a). In the absence of a geologic repository program, the United States has turned to the possibility of extended storage of spent nuclear fuel. Two commercial sites in New Mexico and Texas (see Figure 5.1(b)) are under licensing consideration by the U.S. NRC (2020j). However, both states' governors have already registered their objections to this proposal. At the direction of Congress, on December 1, 2021, the U.S. Department of Energy (DOE) initiated a request for information on the use of consent-based siting of a consolidated storage facility for spent nuclear fuel (DOE-NE, 2021h).

5.2.2.1 Evolution of Spent Fuel Over Time

Notably, spent nuclear fuel is not a passive material, even as it might be stored at centrally located facilities over long periods; rather, its composition and properties evolve over time because of radioactive decay (Peterson and Wagner, 2014). The thermal output and the type and intensity of the radiation field change significantly over time (Hedin, 1997). As an example, because of changes in isotopic composition, the activity level (i.e., radioactive decay rate) of the fuel increases for a short time after discharge (from decay of short-lived fission product neutron absorbers [e.g., xenon-135]) and then decreases out to ~100 years. There is a second peak in reactivity at about 30,000 years caused by radioactive decay of actinide neutron absorbers (e.g., americium-241 and plutonium-240). Hence, the activity or decay rate of the fuel is important over periods associated with interim storage, as well as those for geologic disposal.

The high level of radioactivity of spent fuel is important to the concept of the fuel being "self-protecting"—that is, too radioactive to handle by a bad actor intent on processing the fuel to reclaim fissile material. The high level of radioactivity is a barrier to handling the fuel because radiation shielding would be required. However, the decrease in the level of radioactivity affects this "self-protecting" strategy. The self-protection is significantly decreased between 70 and 120 years after discharge from the reactor. Because many spent fuel assemblies have already been in storage for decades, this self-protection could be lost for some fuel assemblies in as little as 30 years (Peterson and Wagner, 2014).

Finally, the evolution of the fuel composition, temperature, and radiation fields over time (the latter of which is depicted in Figure 5.2) will have a huge effect on the strategy for geologic disposal. In general, cooler fuel will be more easily handled in a geologic repository; however, the evolving radiation field (e.g., the persistence of alpha decay) may lead to radiolytic decomposition of water and accelerate the corrosion rate of the uranium dioxide in spent nuclear fuel. The complex evolution of spent fuel in a disposal environment is described by Ewing (2015) (see Figure 5.3). Hence, the changes in the properties of spent fuel have an impact on both interim storage and geologic disposal strategies.

While Figures 5.2 and 5.3 illustrate these changes for spent uranium dioxide fuel, as one envisions the behavior of new types of fuels in the disposal environment, the same processes must be considered. New fuel types, such as TRistructural ISOtropic (TRISO) fuel, may increase the complexity of the analysis but also may show better performance, such as containment of fission products, because of the silicon carbide layer surrounding the TRISO kernels.

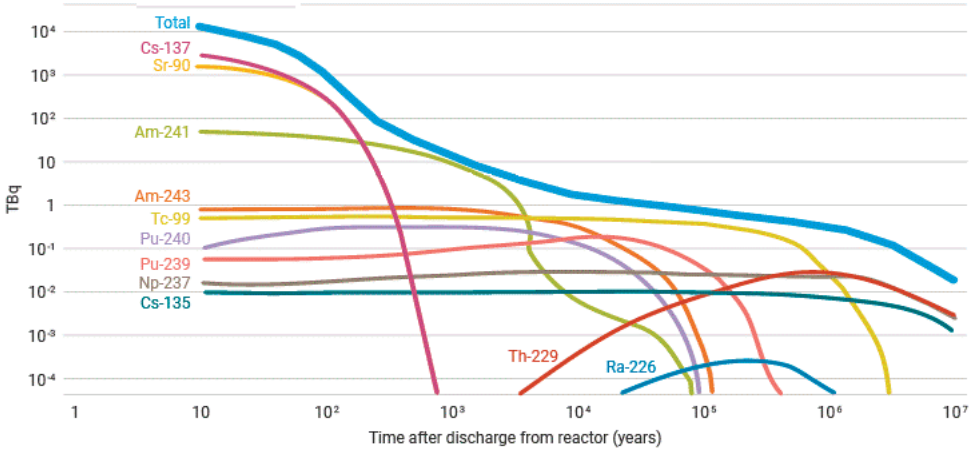


FIGURE 5.2 Radioactivity of one metric ton of spent nuclear fuel over time.
NOTE: TBq = terabecquerel.
SOURCE: IAEA (1982) *Radioactive Waste Management*, adapted by WNA (2022d). Reproduced by permission of the International Atomic Energy Agency, "Nuclear Power, the Environment and Man," IAEA, Vienna (1982).

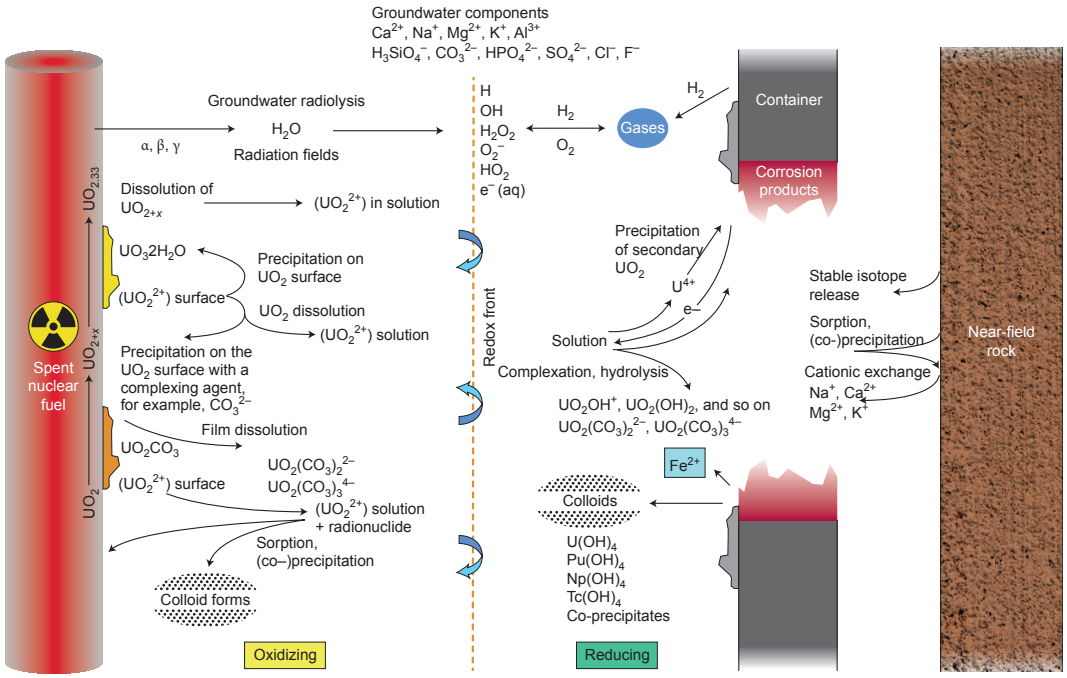


FIGURE 5.3 Evolution of spent fuel in a geologic disposal environment. This schematic diagram illustrates the processes important to the description of the corrosion of UO_2 in spent nuclear fuel. These include dissolution of the fuel; precipitation of secondary phases onto the surface of the fuel; colloid formation; sorption of radionuclides onto corrosion products of the metal canister; formation of gases such as H_2 during corrosion of metal; formation of chemical species in solution; cation exchange/sorption reactions between radionuclides and host rock; and the formation H_2O_2, O_2, and OH due to radiolysis. The rates of all of these processes are determined by the changing thermal gradient between the fuel and the host rock, level of radioactivity, and flow rate of the groundwater.
SOURCES: Ewing (2015); Grambow et al. (2000). © European Communities, 2000.

Although other countries also have struggled with geologic disposal, and almost all have had to stop and reorient their approach, some are making good progress, including Sweden, Finland, France, Canada, and Switzerland. In Sweden, Finland, and France, which have selected repository sites, implementers have moved forward in their applications for approval from regulatory authorities. In Finland, the geologic repository is already under construction, with the expectation of receiving the first spent fuel for disposal around 2025. In Sweden, as this report was being written, the government approved the construction of a repository near Östhammar. In a review of international experience in site selection and the development of a mined, geologic repository, the Nuclear Waste Technical Review Board in 2015 analyzed the history of 24 efforts to site geologic repositories in 10 countries (NWTRB, 2015). Of these national programs, only 6 have remained on track. Most importantly, all programs have experienced a point at which it was necessary to completely reorient their program and essentially begin again. *The waste management and disposal program of the United States is at such a point of reckoning.*

5.2.3 Costs of Waste Management

For some 20 years, the United States investigated the Yucca Mountain site as a potential geologic repository. As of 2009, this effort had cost $15 billion (GAO, 2011), and in 2008, DOE estimated the total system life-cycle cost of the Yucca Mountain project to be $96.2 billion (2007$) (DOE, 2008). In addition, the cost of stranded waste and unfinished facilities has been substantial. Two notable examples are the mixed oxide fuel fabrication facility at the Savannah River Site and the vitrification plant at Hanford. Any plan to reprocess spent fuel from advanced reactors is likely to entail (1) the fabrication of nuclear fuels that are a mixture of uranium and plutonium and (2) the solidification of high-level waste from reprocessing prior to disposal. In 2000, a contract was awarded for the Hanford Waste Treatment Plant with a projected cost of $4.3 billion and a startup in 2007. In 2006, the cost estimate was revised to $12.3 billion, and the startup was delayed to 2019; it has been further delayed to the 2030s. In 2020, the U.S. Government Accountability Office (GAO) documented this history and noted DOE's "2019 Hanford Lifecycle Scope, Schedule and Cost Report that completing the WTP [waste treatment plant] would cost between $19 billion and $30 billion, in addition to the more than $11 billion already spent" (GAO, 2020).

In 2007, at the Savannah River Site, work began on the mixed oxide fuel fabrication facility for the disposition of 34 MT (metric tons) of plutonium from dismantled nuclear weapons, with an initial cost estimate of $4.9 billion. In 2012, GAO estimated that the cost had risen to $7.7 billion with the earliest start-up date being 2019 (GAO, 2013b). Estimates of the life-cycle cost reached as high as $30 billion (Holt and Nikitin, 2017). Despite early interest in using the mixed oxide fuel, no utility finally agreed to use it. The construction of the mixed oxide fuel fabrication facility ended in 2018 even though major portions had already been completed (GAO, 2019). *To the extent that advanced reactors are heralded as a first step toward a closed fuel cycle in the United States, one needs to carefully consider the cost and experience with the construction of facilities that will be required to support the advanced fuel cycles proposed.*

Finally, because of the failure of DOE to take ownership in January 1998 of the spent nuclear fuel generated and stored at reactor sites across the nation, utilities have successfully sued DOE and collected several hundred million dollars annually from the Judgment Fund (i.e., the U.S. taxpayer) for the costs related to continued storage of spent nuclear fuel at reactor sites. Through fiscal year (FY) 2020, payments to utilities from this fund totaled $8.6 billion. DOE has estimated "that its potential liabilities for waste program delays could total as much as $39.2 billion" (Holt, 2021a).

These examples illustrate the high costs and long delays that result from failing to plan for all the necessary facilities to support the back end of the nuclear fuel cycle—whether open or closed.

5.2.4 U.S. Nuclear Waste Management Today

At present, the back end of the U.S. nuclear fuel cycle is a jumble of different facilities and activities that have resulted from poor planning, continued programmatic delays, and the failure to create a geologic repository.

How Did This Happen?

Broadly speaking, the causes were (1) changes to the original NWPA of 1982; (2) a slowly developing and changing regulatory framework; (3) erratic funding; (4) consequential changes in policy with changing administrations; (5) conflicting congressional and federal agency policies; and (6) insufficient public engagement in decisions about fundamental strategies for the storage and disposal of highly radioactive nuclear waste (Reset, 2018).

More specifically, the grand covenant hammered out by the interagency process prior to and incorporated into the NWPA of 1982 has unraveled. In particular,

- sites for a repository were to have been chosen by a technically driven evaluation of at least three candidate sites—the result of a down selection process from a greater number of sites;
- in order to promote geographic equity, two repositories were to have been developed;
- a fee levied on the use of electricity from nuclear power plants was to have funded the development of a geologic repository so that the federal government could take ownership of the spent fuel by January 31, 1998; and
- states were to have been given a meaningful role in the approval of the selected site.

In contrast to expectations, Congress designated the Yucca Mountain site as the only site for characterization in the Nuclear Policy Act Amendments of 1987; the goal of a second repository, presumably in the eastern United States, was not pursued; the Nuclear Waste Fund ratepayer money has grown to more than $40 billion, but has not been generally available for the development of a geologic repository; states have had a limited role in site selection; and most notably, the state of Nevada has waged a tenacious battle against the Yucca Mountain site as the nation's geologic repository for more than 35 years. Finally, the Obama administration eliminated the Office of Civilian Radioactive Waste Management, and Congress has not funded the Yucca Mountain project since 2010. Presently, the United States continues to have no strategy for the management and disposal of spent fuel generated by commercial nuclear power plants during the past or into the future.

In addition to this broad backdrop of major challenges, the United States also faces important limitations in its ability to address nuclear issues. As an example, the Idaho National Laboratory (INL) is the lead laboratory for nuclear energy research. However, in 1995 the state of Idaho reached an agreement with DOE that all spent fuel on site would be removed by 2035 (IDEQ, 1995). This agreement has interfered with research that requires that spent fuel be brought to INL for research purposes. In the absence of a geologic repository, the prospects of moving spent fuel out of Idaho by 2035 dim with each passing year.

Another major issue is one of human capital and research support required for researchers to develop appropriate skills related to all aspects of nuclear fuel. More specifically, the Yucca Mountain project has been on hold or dead for more than a decade, and during that time, many researchers have moved to other subjects or retired. The loss of this knowledge base will certainly delay the success of programs that support the back end of the nuclear fuel cycle.

Why is this story important for the potential development of alternative fuel cycles? As the United States looks forward to any expansion of nuclear power, there should be a plan for the disposal of the spent nuclear fuel and other radioactive wastes that result from the use of advanced reactors and small modular reactors. *The 40-year history of delay, rising costs, and finally failure to deal with highly radioactive waste is a legacy that this generation should not pass on to future generations.*

5.2.4.1 The Way Forward

During the past decade, a number of groups have reviewed the U.S. situation and made recommendations that, if followed, would provide a basis for a new U.S. program for nuclear waste management and disposal. The most prominent effort was undertaken by the Blue Ribbon Commission on America's Nuclear Future (BRC, 2012).

Their final report was submitted to Secretary of Energy Steven Chu in January 2012, and its recommendations focused on nuclear waste management issues. Relevant, principal recommendations included

- The urgent need for one or more geologic repositories;
- The formation of a single-purpose organization to implement an integrated program of transportation, storage, and disposal of nuclear waste;
- assured and continued access to the Nuclear Waste Fund, which now has a balance of more than $40 billion; and
- the use of an adaptive, staged, transparent, and particularly consent-based process in the siting of a geologic repository.

In 2018, a 3-year study, Reset of America's Nuclear Waste Management Strategy and Policy, was completed at Stanford University and The George Washington University (Reset, 2018). The study, guided by a steering committee of nuclear waste experts, sought to identify the systemic issues that have prevented the success of the U.S. program. Key issues identified include

- the need for a new, independent, single-purpose, not-for-profit national radioactive waste management and disposal organization, and reform of the funding process;
- integration of the back end of the nuclear fuel cycle, such that all decisions are focused on the common, primary goal of geologic disposal of nuclear waste;
- a new approach to public engagement that develops a consent-based siting process based on trust in the implementer, redistribution of power among affected parties, and defined roles in the decision-making process; and
- a reexamination of the regulatory framework with a focus on a safety-case approach using both quantitative and qualitative criteria.

The linchpin of both sets of recommendations is the creation of a new waste management organization. The Blue Ribbon Commission recommended a federal corporation similar to the Tennessee Valley Authority, while the Reset committee pointed to the unique advantages of a not-for-profit, utility-owned waste management organization. A utility-owned organization would provide a "cradle-to-grave" approach from the moment the spent fuel is removed from a reactor until it is disposed of in a geologic repository.

Importantly, the recommendation for a new organization is not new. Willrich and Lester in 1977 first recommended the establishment of a "Radioactive Waste Authority," as they had concluded "the existing organization for radioactive waste management is likely to be unworkable if left unchanged" (Willrich and Lester, 1977). And it was. The successful nuclear waste management and disposal programs—those in Finland, Sweden, Canada, and Switzerland, in particular—are managed by a utility-owned organization, not a government agency. *A new organization, as the implementer of a waste management and disposal strategy, would have to be a top priority in any plan to expand nuclear power in the United States.*

The establishment of a new organization would also require careful attention to a new funding approach so that funds could be used in a timely manner over the many decades required to select a site and design, license, and construct a new geologic repository. In the case of a utility-owned organization, this would require the transfer of the Nuclear Waste Fund, over an extended time, to that organization.

In addition to creating a new implementing organization, the funding of the implementer must be fixed, the regulatory framework reexamined, and a robust approach to siting that engages and addresses the concerns of the public must be followed to ensure success in repository siting. Creating a new organization and funding scheme, and developing the structure and values of a new organization would take time, but the basic roadmap has been available for a decade. In fact, DOE is on record as having accepted all of the recommendations of the Blue Ribbon Commission (DOE, 2013). Hence, there is a way forward, but this will require congressional action.

5.3 THE CONCEPT OF GEOLOGIC DISPOSAL OF HIGHLY RADIOACTIVE NUCLEAR WASTE

Some advocates for advanced reactors and small modular reactors claim that one benefit of these technologies would be to ease the burden on nuclear waste disposal strategies. A key question for this committee is whether these reactors could be a "silver bullet" for what has been a vexing, national problem. In order to evaluate the veracity of such claims, key aspects of the concept of geologic disposal[3] are first summarized.

5.3.1 The Concept of Geologic Disposal

Many variations in the strategies for geologic disposal become evident in the comparison of repositories in different geologic settings; however, all approaches share the common strategy of using multiple barriers in order to provide defense in depth. In the face of the uncertainty in the extrapolated behavior of a repository over long periods, a system of multiple barriers, also known as the "Russian-doll" or "belt-and-suspenders" approach, provides a robust basis for confidence in the ability of a geologic repository to contain or reduce the mobility of key radionuclides, thereby lowering radionuclide access to the environment and subsequent exposure to humans (Ewing and Park, 2021; Hedin, 1997). Descriptions of the two types of barriers follow.

Engineered barriers include the spent fuel or nuclear waste form, the waste package, backfill and/or overpack, and structures to prevent the ingress of water to the waste package and/or the egress of radionuclides from corroded canisters. Most engineered barriers are physical, but in some cases, they are chemical additions designed to reduce radionuclide concentrations, such as the magnesium oxide emplaced with the transuranic waste in the disposal panels at the Waste Isolation Pilot Plant, a geologic repository for defense transuranic waste located in southeastern New Mexico.

Geologic barriers rely on the properties of the rock and the hydrologic system of the repository. The petrophysical, geochemical, and hydrologic properties of the rock into which a repository is constructed can have major effects on the mobility and concentrations of radionuclides. Judgment of whether a site is acceptable depends on whether the geologic setting and hydrologic system meet the regulatory requirements of the safety case.

As simple as this approach appears to be, its application is complicated when developing the safety assessment, which can have hundreds of subsystem models and many hundreds of input parameters. Most importantly, the subsystems that comprise individual barriers are highly coupled and nonlinear. Larger-scale coupling leads to the development of codes to capture the thermal-mechanical-chemical-hydrologic regimes, while models of near-field interactions capture the long-term geochemical evolution of waste form–waste package–backfill materials using reactive transport modeling (Arcos et al., 2008). As an example, in the KBS-3 waste disposal concept developed in Sweden, experiments and modeled data show the impact of accessory minerals and clay surfaces on pore water chemistry. The pore water chemistry affects the long-term stability of the bentonite barrier.

Thus, any aspect of the analysis that can be simplified by, say, a very durable waste form or very long-lived waste package can make for a simpler and more compelling safety case. As an example, if an advanced reactor uses fuel that is unusually durable in the disposal environment, this could represent an important contribution to safety; however, the variation in hydrogeochemical conditions present in a repository will necessarily cause differences in the expected behavior of the fuel due to coupled chemical reactions in an evolving thermal and radiation field.

5.3.2 Different Strategies for Geologic Disposal, or Not Every Hole in the Ground Is the Same

During the past 30 years, considerable research has been conducted on scientific and engineering issues that support geologic disposal. Importantly, different countries have pursued the development of geologic repositories in different geologic settings that can be described most simply in terms of their rock type (Ewing and Park, 2021): salt, clay, crystalline rock, and volcanic tuff. Even a cursory examination of the different national programs in a variety of geologic settings shows that the fundamental safety strategies vary as they are adapted to the specific waste types and geologic settings. In some instances, the engineered barriers are most important (e.g., corrosion

[3] For a tutorial on the concept of geologic disposal, see Piet Zuidema's presentation to the committee (Zuidema, 2021), particularly the figures on slides 28, 35, 36, and 46.

resistant copper canisters), and in other cases, the geologic barrier plays the critical role (e.g., plastic deformation and sealing by salt or clay). Ideally, all of the barriers will play an important role in containing radionuclides, generally over different time frames as the composition of the waste and the conditions in the repository evolve with time (Ewing, 1995).

This report does not provide a detailed review of the different approaches for geologic disposal; rather, the committee refers the reader to Yardley et al. (2016) and NWTRB (2015) for summaries of the approaches taken for each of the different geologic settings. Importantly, major differences in basic strategies result from the following:

- The thermal pulse from short-lived fission products (e.g., aging the waste prior to disposal versus the use of smaller waste packages). The longer-term thermal pulse from actinides is of less importance to repository performance. However, the alpha decay of actinides is important because of the effect of the radiolysis of water at the surface of the fuel and the potential for changing the redox conditions from reducing to oxidizing.
- The release of long-lived, highly mobile fission and activation product elements (e.g., by relying on containment in packages of exceptional durability or dilution during release and transport).
- The release of long-lived actinides that make up the bulk of the chemistry for spent fuel disposal (e.g., by disposal in a reducing environment versus an oxidizing environment). The redox state of the repository can profoundly affect the mobility of the multivalent actinides.

These three examples, listed in order of increasing half-life, highlight the major challenges associated with the composition of the waste. For the short-lived radionuclides, aging, ventilation on the emplacement, or changes in waste package size enable effective engineering solutions. For long-lived, highly mobile fission and activation products, the challenge is greater because a number of these elements travel in groundwater as negatively charged species; hence, they are not sorbed onto mineral surfaces, which are also negatively charged. Possible solutions to these problems can be as simple as dilution, particularly isotopic dilution of iodine, or incorporation into very durable waste forms, such as iodine into apatite (Ewing and Wang, 2002). For long-lived actinides, with half-lives on the scale of many hundreds of million years, ensuring geochemical (e.g., reducing conditions) and hydrologic conditions (e.g., transport by diffusion) that drastically reduce their mobility in the repository environment is important. As an example, disposal of actinides in the reducing conditions of an organic clay can essentially limit the mobility of actinides to within some tens of meters from the point of emplacement for a million years (Grambow, 2008).

Recently, there has been increased interest in the potential for deep borehole disposal for specific waste types, including spent nuclear fuel and high-level waste generated by advanced reactors. Present efforts in the United States are supported by two private companies, Deep Isolation, Inc., and NuclearSAFE (Deep Isolation, 2022; NuclearSAFE, 2018). Both companies propose to use deep boreholes combined with directional drilling for horizontal emplacement of nuclear waste–containing packages. The deep borehole disposal concept was first considered in the 1970s, but the concept was not pursued because drilling technology was not sufficiently advanced at the time. Recent renewed interest in the concept reflects the rapid advancement of drilling technologies from the oil and gas industry. Deep borehole disposal concepts rely on the isolation capacity of the geosphere and the deep hydrologic environment. A recent study by the Electric Power Research Institute (EPRI) (2020) on the feasibility of deep boreholes collocated with advanced reactors examined key elements such as site characterization, regulatory considerations, spent fuel and waste package characteristics, and operations and safety analysis; this report did not identify any technical showstoppers.[4] Another study (Krall et al., 2020) examined and identified technological challenges of siting deep boreholes. Since the details of potential waste characteristics of many advanced reactors are not yet available, the committee decided not to examine the concept of deep borehole disposal at this point. But as advanced reactors progress and mature, an assessment of the deep borehole concept may be warranted.[5]

[4] The committee notes that Deep Isolation, Inc., was a contractor for the EPRI report.

[5] One committee member disagreed with this statement, given the technical challenges that have been identified for deep borehole disposal in independent analyses such as Krall et al. (2020) and references therein.

5.3.3 Characteristics of a Geologic Repository

Although the basic physics of all fission-based nuclear power plants is the same, a wide variety of designs is possible, depending on the type and enrichment of the fuel, the composition of the moderator, and the type of coolant, to name a few salient features. Thus, advanced reactor designs reveal a broad array of strategies for producing energy from fission reactions.

In much the same way, the basic physiochemical processes in the natural and engineered environments of a geologic repository are similar, but the basic strategies for containment of radionuclides may differ in important ways, depending on the type of fuel, level of burnup, type and degree of reliance on engineered barriers, and properties of the geologic setting. This brief section characterizes some of the major differences in repository strategies. *The goal of geologic disposal should be to match the different types of waste generated by different reactor types with the most appropriate disposal strategy.*

The simplest characterization of the geologic setting is usually given by noting the rock type: salt (e.g., dome or bedded formation), crystalline rock (e.g., igneous granite or metamorphic gneiss), clay and shale, and volcanic tuff. However, categorization by rock type fails to capture important differences in geologic settings. As an example, salt domes have very low water content (only fluid inclusions); bedded salt deposits have a greater abundance of fluid inclusions and the possibility of large, pressurized brine pockets; shales contain water in the atomic structure of clays, and water flow is hindered so much that in "tight" shales, radionuclides migrate mainly by diffusion; crystalline rock contains water in fracture systems that move radionuclides by advective flow; and in volcanic tuff, substantial amounts of water are typically held within the relatively high porosity (10 to 15 percent), and this water is mobilized at elevated temperatures. Hence, the properties specific to each geologic setting will drive the development of different waste disposal strategies. These might include upper limits on the heat load to reduce chemical reaction rates, preserve the chemical properties of clay rock or bentonite backfill, or ensure redox conditions that chemically stabilize key radionuclides (mainly the actinides). Reducing redox conditions are generally considered important; hence, most repositories are designed to maintain reducing conditions. Importantly, for uranium, reducing conditions lower its solubility by nearly four orders of magnitude (Shoesmith, 2000).

Thus, the final strategies will be driven by not only the type of waste disposed of but also the petrophysical, petrochemical, and hydrologic properties of the specific geologic setting (see Table 5.2). Because in some countries it is difficult to change the geologic setting, there will be greater emphasis on the engineered barriers. Even in strategies that rely mainly on engineered barriers, geologic barriers remain an important aspect of the safety case.

The strategy also depends on the regulatory framework. The goal may be complete containment within a short distance of the repository, or the point of compliance may be tens of kilometers away from the repository. In the latter case, sorption and dilution are important factors in the demonstration of compliance, even though they are well beyond the repository itself. Thus, it is not always appropriate to compare the safety assessment of one repository type with another, as the determination of safety may be based on very different compliance distances and time frames.

Finally, although strategies may vary, initially most of the radionuclides are contained within the spent fuel or, in the case of reprocessing, the waste form selected for reprocessed waste. Hence, the durability of the spent fuel or nuclear waste form is the first and last barrier to radionuclide release. This means that any selected strategy may focus attention on the properties of the fuel. *New types of fuel will require research programs that substantiate their compatibility and durability in the expected disposal environment*. Such research programs require considerable attention to a wide variety of processes, including redox conditions, radiolysis of water, solubility limits, and colloid formation, to name a few (Ewing, 2015).

5.3.4 Grand Challenges of Geologic Disposal

In the discussion of advanced reactors and advanced fuel cycles, there are often claims that the new reactors, perhaps combined with chemical processing, will improve the ability of the United States to manage and dispose of its nuclear waste. In order to evaluate such claims, it is necessary to understand both the progress that has been made and the challenges remaining for nuclear waste management and disposal of highly radioactive waste.

TABLE 5.2 Properties of Different Host Rock Types Considered for Geologic Disposal

Host Rock	Safety Properties Associated with the Host Rock and Natural Environment	Safety Concerns Associated with the Host Rock	Importance of Engineered Barriers	Countries Committed to the Concept	Countries Actively Investigating the Concept
Salt	• Absence of flowing water • Self-healing fractures • High thermal conductivity to remove heat	• Heat induces moisture movement • Hydrogen gas buildup • Increased likelihood of human intrusion for natural resources • Corrosivity of any intruding water	• High at WIPP (magnesium oxide to protect against the consequences of human intrusion)		• Germany • United States
Crystalline rock	• Stable for mining • Provides compatible environment for engineered barriers • Low fracture density	• Corrosion of metal canister • Stability of bentonite buffer • Changes to the geohydrological and geochemical conditions	• High (e.g., copper canisters and bentonite clay)	• Finland • Sweden	• Canada • China • Japan • United Kingdom • United States
Clay/shale	• Self-sealing fractures • Diffusion-controlled radionuclide migration • High sorption capacity	• Potential for permeable faults • Increased likelihood of human intrusion for natural resources	• High (vitrified waste forms and/or corrosion-resistant waste packages)	• Belgium • France • Switzerland	• Canada • China • Japan • United Kingdom • United States
Volcanic tuff at Yucca Mountain, Nevada	• Arid climate reduces the amount of water entering the repository drifts • Closed hydrologic basin limits the distance that radionuclides can travel	• Uncertainty about the presence of fast flow paths • Potential for deliquescence-induced corrosion of the waste package • Oxidizing conditions, which allow for mobilization of radionuclides • Heat-induced moisture movement	• High (corrosion-resistant waste packages and drip shields)	• United States (currently in political and legal limbo)	

NOTE: WIPP = Waste Isolation Pilot Plant.
SOURCE: Reproduced from NWTRB (2015).

The grand challenges are as follows:

1. *The investigation of multiple sites and the final selection of a single site for a geologic repository.* There are multiple approaches to repository siting, but successful siting depends on finding sites that are both technically and societally acceptable. On the technical side, a major issue has been establishing appropriate criteria against which sites can be evaluated and compared. The criteria are of three types: (a) exclusionary, (b) host rock specific, or (c) generic (Ewing and Park, 2021; NWTRB, 2015). Arriving at an early consensus on the appropriate criteria greatly simplifies the evaluation and comparison of sites. However, coming to agreement on these criteria, particularly when comparing different geologic settings, remains difficult.
2. *The social process of siting a repository.* In many countries, the institutions responsible for developing a geologic repository have appreciated that the process of siting a geologic repository is not only a technical process, but equally a social process that requires extensive and prolonged public engagement and empowerment (Metlay, 2016, 2021). The United States began to accept this understanding with the report of the Blue Ribbon Commission, which emphasized the need for consent-based siting processes, but the country is still far from ready to implement such a process.
3. *The chemical and radiological complexity of the nuclear waste streams*. The fission process creates the following types of waste radionuclides: (a) short-lived fission products (e.g., iodine-131, cesium-137, and strontium-90), which generate intense radiation fields and considerable heat; (b) long-lived fission products (e.g., iodine-129, technetium-99, selenium-79); (c) long-lived actinides (e.g., uranium-235, uranium-238, plutonium-239); (d) minor actinides (e.g., isotopes of americium and curium); and (e) activation products (e.g., cobalt-60, nickel-63). Each of these radionuclides has its own chemistry and hence different mobilities in the natural environment.

   Site selection can play an important role in determining the mobility of radionuclides, as the geologic setting and related fluid compositions and hydrologic flow determine radionuclide concentrations in solution and the distances over which they may be transported. In addition, the thermal output and radiation fields (type and strength) will change with radioactive decay (Hedin, 1997). Hence, the composition of the waste will evolve because of radioactive decay, and the changing conditions induced by the changing radiation and thermal fields will impact conditions within the repository. Also, the operational parameters of nuclear power plants or chemical processing facilities and related off-gassing will generate considerable low-level waste, including GTCC waste, some of which will require geologic disposal or near-surface disposal, depending on the radionuclides contained. In the broadest sense, advanced reactors and their associated fuel cycles (e.g., reprocessing versus no reprocessing) may impact the compositional complexity of the highly radioactive waste streams.
4. *Modeling behavior of radionuclides in a geologic repository over extended periods that stretch to hundreds of thousands of years and in some cases to a few million years*. From a technical perspective, this is a "wicked" problem further complicated by public fear of radiation and the need to make credible and compelling extrapolations of long-term behavior.
5. *The proper use of models for the extrapolated behavior of the repository.* A safety assessment can include models ranging from atomic-scale behavior of corroding spent fuel to crustal scale models of tectonics and hydrology, extending over many tens of square kilometers. At every step of a safety assessment, there are models of physical, chemical, and biological processes, as well as human behavior that affect exposure rates (Swift and Sassani, 2019). The role of modeled performance and its relation to safety assessment remains much debated.
6. *Safety considerations*. This entails developing a clear understanding of the meaning of *safety* in the context of calculated risk and regulations. The issue here is that there are many metrics for the evaluation of the safety of a repository. Typical metrics are the composition of the waste, amount and types of radioactivity, radiotoxicity, thermal output, distribution of fissile materials, volume of disposed waste, mobility of released radionuclides, and calculations of exposure to groups and individuals over extended timescales that stretch to hundreds of thousands of years. These metrics provide a variety of ways to compare advanced

fuel cycles with "traditional" LWR fuel cycles; however, these metrics are not equally appropriate or useful when comparing different geologic disposal strategies.

7. *Regulatory framework*. Today, the United States has no viable regulatory basis for licensing a geologic repository for spent nuclear fuel, except at the Yucca Mountain site, because the regulatory framework in the United States is site specific. Creating the framework for licensing a geologic repository began in 1980 (10 CFR 60) and evolved over 40 years to the present predicament. The coordinated efforts of three federal agencies; extensive engagement with the public, states, and tribal governments; and a constant stream of litigation, often requiring federal agencies to amend rulemaking and requirements, were responsible for the long time required (Ewing, 2011). Appendix D provides information on the development of waste classifications and their jurisdictional bases.

   Any new repository site, even if sited in a willing community, will require licensing according to standards and regulations. The new standards and regulations may be site specific or generic. In any case, one can expect the process to take decades. While it is possible to develop general guidelines and standards to facilitate the siting of a new repository, there is no federal effort or organization in place to address this, casting doubt on the country's ability to fully dispose of nuclear waste from closed or open fuel cycles.

Of these seven grand challenges for nuclear waste disposal, only the third may be impacted directly by advanced reactors and their fuel cycles. Two contributions could be made by advanced reactor fuel cycles to the safety case for a geologic repository: (1) changing the composition of waste streams for disposal or (2) improving the durability of nuclear fuels and nuclear waste forms in the disposal environment (Peters and Ewing, 2007). As one looks to advanced reactors and their fuel cycles, particularly with reprocessing, proponents need to clearly articulate the benefits to addressing the challenges of geologic disposal.

What can already be concluded is that all advanced fuel cycles will require a geologic repository. A new U.S. repository will require many decades of effort before the repository can accept nuclear waste for disposal.

Additionally, advanced reactors that change the inventory of actinide isotopes by using new fuel types, increased levels of enrichment, higher burnup, or use of actinides from reprocessing will not be solving an intractable waste disposal problem because the issue of the mobility of actinides can be addressed by the thoughtful selection of a site that has reducing conditions.

5.4 WHAT MATTERS?: WHAT THE COMMITTEE LEARNED FROM THE EXPERTS

During its deliberations, the committee was privileged to hear from a number of national and international experts on nuclear waste management and disposal. (See Appendix B for a listing of the committee's public information-gathering sessions.) The following discussion outlines some important, specific points made during these presentations.

5.4.1 Important Factors for Evaluating the Safety of Geologic Repositories

The most common safety indicators for evaluating the impact of advanced reactors on the safety of a geologic repository include radiotoxicity inventory, radiotoxicity isolation, mobile fraction of radionuclide inventory, maximum endpoint dose, and comparison of released concentrations with natural occurrences of uranium (Grambow, 2021). For new fuel types being proposed that will use HALEU, another factor to examine in relation to repository safety is criticality. A careful analysis of each of these parameters reveals their limitations and emphasizes that the main issue is the interplay of toxicity and mobility in the geologic environment (Grambow, 2021). As an example, repositories in Sweden and Finland, under reducing conditions, show a peak dose that is governed by mobile radium-226, rather than plutonium, which has a much lower mobility in the granitic geologic setting. In contrast, the dose estimates for the proposed repository at Yucca Mountain, under oxidizing conditions, are dominated by plutonium contribution to risk assessments because of the higher mobility of plutonium under oxidizing conditions. Because the assessment of the fate of plutonium is estimated by solubility limits and sorption of the radionuclides at Yucca Mountain, the inventory reduction of actinides has little or no effect on disposal risk (Grambow, 2021).

Alternative fuel cycles may lead to the development of new waste forms, but the final analysis evaluating

the safety of a geologic repository depends on the repository concept and the efficacy of the multibarriers system. For example, for spent uranium oxide fuel on a mass basis of an individual fuel assembly,[6] higher burnups lead to both higher inventories of fission products and faster release of iodine-129 (Grambow, 2021). However, the release of the most radiotoxic nuclides, such as isotopes of plutonium, depends on whether the geologic setting is oxidizing or reducing. The performance of mixed oxide fuel is similar to that of uranium oxide fuel, but the release of iodine-129 is faster than that of uranium oxide fuel, and the heat load will exceed that of spent uranium oxide fuel by as much as 33 percent. For vitrified waste from reprocessing, borosilicate glass is not an effective waste form for the retention of iodine-129. *In the evaluation of repository performance and safety, the fuel cycle is less important than the redox state of the geologic repository.*

5.4.2 Impact of Wastes from Advanced Reactors and Fuel Cycles on Geologic Disposal

The amounts and types of waste that will be generated by advanced reactors are difficult to estimate at this early stage of the development of advanced reactors; yet, this type of information is required in order to determine the impact of advanced reactors and advanced fuel cycles on the back end of the fuel cycle. Actual advanced fuel cycle options may reduce resource consumption or increase the energy output, but they are unlikely to reduce the long-term risks of waste management and disposal (Grambow, 2021). In most geologic settings, the dominant contributors to long-term dose estimates are mobile species (e.g., iodine-129), so the impact of alternative fuel cycles on estimates of long-term repository performance will be minimal (Swift, 2021; Swift and Sassani, 2019). Additionally, disposal volumes depend less on the waste volume generated for disposal and more on the heat generation from the wastes (Grambow, 2021), and without century-scale aging of fission products, alternative fuel cycles will not significantly impact thermal load management (Swift, 2021; Swift and Sassani, 2019). And, regardless, regulatory criteria may necessitate the disposal of fission products in a geologic repository (Swift, 2021; Swift and Sassani, 2019).

5.4.3 Additional Factors Influencing Fuel Cycle Decisions

As discussed in Chapter 2, the French nuclear program has established a long-term strategy for research, demonstration, and deployment of advanced reactors and fuel cycles. Therefore, insights and decision-making strategies from the French program provided useful information to this committee. First, because natural uranium resources are still available at low cost, the French nuclear industry is focused, in the short and medium time frames, on the success of third-generation reactors and does not plan to demonstrate and deploy fast breeder reactors until at least the second half of the 21st century (Landais, 2021). The public's impression of nuclear power is "mostly determined by the accidents that have marked its history," and emerging technologies, such as small modular reactors and fast-neutron reactors, do not at this stage change the public image of the nuclear industry in France (Landais, 2021). Furthermore, the role of nuclear energy in reducing greenhouse gas emissions continues to be highly debated within the European Community. As an example, the European Commission (EC) will issue a €250 billion "green bond," but until 2021 the European Union excluded the use of green bonds for nuclear power plant projects (Landais, 2021). In a recent decision, the EC has included nuclear and natural gas as "green bond" projects, and in July 2022 the EU parliament voted in support of labeling nuclear and natural gas as green investments, with regulations expected in early 2023 (Ainger, 2022). Finally, the French emphasize that a key element of a successful fuel cycle, even one that includes reprocessing, is the implementation of a deep geologic repository program (Landais, 2021).

Based on the presentations and committee member expertise, the committee has concluded that the introduction and use of advanced reactors and small modular reactors in and of themselves will do little, if anything, to eliminate the need to manage and dispose of nuclear waste. Even if advanced fuel cycles were developed and implemented that reduced the actinide inventory 1,000-fold or more, the short- and long-lived fission products would still require management and isolation from the biosphere (geosphere) for long periods of time. Indeed, in some cases the amounts of certain categories of waste generated may increase, and certainly the types and characteristics of

[6]This sentence has been modified from a prepublication version of the report to clarify the statement on spent uranium oxide fuel.

the highly radioactive waste and other waste categories—such as TRISO-based fuels, more highly enriched fuels, and higher-burnup fuels—may require focused research programs prior to their disposal in a geologic repository. That said, careful attention to the types of fuels and waste that require disposal and the selection of appropriate geologic settings should lead to the safe disposal of these fuels.

Also deserving careful attention is the combination of advanced reactors with different reprocessing schemes for actinides in fuel cycles that involve recycling: monorecycling, multiple recycling to close the fuel cycle, or different recycling schemes for managing the fate of minor actinides (Poinssot and Boullis, 2012). These factors materially impact the challenges of geologic disposal. Although each of these recycling schemes may reduce the actinide content of the waste, particularly for transuranic elements, they may actually increase the actinide content in the fuel cycle. Still, the long-term safety of disposal of actinides in appropriate geologic settings is assured in most geologic disposal projects largely independent of the actinide inventory of the repository (Grambow, 2008). *The main challenge is the containment of long-lived fission product elements that, due to their chemistry and mobility in most geologic environments, can be major contributors to radiation dose over long periods*. The main advantage of actinide recycling is the conservation of uranium resources at the front end of the fuel cycle and the associated reduction in the environmental impact of uranium mining.

5.5 SPECIFIC WASTE ISSUES THAT ARISE FROM ADVANCED NUCLEAR REACTORS AND FUEL CYCLES

Many advanced nuclear reactor concepts have been proposed in recent years, most of which use nonwater coolants and innovative fuels, with graphite moderators for thermal reactors. Many of these advanced reactors also use uranium fuels with higher enrichment than what is used in the current pressurized water reactor (PWR) fleet and produce high-burnup spent fuels. These advanced reactors are currently at various stages of design and development. Notably, none has reached the operating demonstration plant stage yet. As a result, it is difficult to be precise about the chemical and physical nature of the irradiated fuels or the details of operational and decommissioning wastes. Nevertheless, it is possible to describe these waste streams in broad chemical and physical terms and identify possible processing, treatment, and disposition pathways based on available information and existing experiences in waste management technologies.

Key characteristics of spent nuclear fuel from these advanced reactors, such as amount, chemical composition, radionuclide inventory, thermal power, and waste form behavior, can in general be estimated using the key design features of the reactor and fuel cycle (Wigeland et al., 2014). However, without detailed designs and operational information, it is more difficult to estimate the types, amounts, and characteristics of low-level waste, including GTCC, generated during operations. As a result, to assess waste generation and management from advanced nuclear reactors, the committee focuses on new and or/unique waste streams generated from the advanced reactors and analyzes their potential advantages and disadvantages to waste management concerns.

Chapter 3 includes detailed descriptions of advanced reactors. As can be seen from Table 3.1, a variety of advanced reactor types are proposed, with different coolants, neutron spectra, fuel types, reactor sizes, and design details. Given the large number of reactor types and the immature stages of development of these reactors, the committee has decided to select a subset of the advanced reactors to analyze their waste management issues. The analyses below are only representative of potential waste management issues.

5.5.1 Waste from iPWRs

Integral pressurized water reactors (iPWRs) are designed with more passive safety features than existing large PWRs and are sized much smaller (notionally less than 300 MWe [megawatts electric]), and they can be used in tandem to create a larger power plant, depending on the needs of the owner. Although iPWRs may experience more neutron leakage than large PWRs, which can negatively impact fuel utilization, they can be designed to produce comparable spent nuclear fuel inventories on a per thermal power basis. For example, NuScale indicates that its iPWR, a 12-module, 924-MWe plant (3,000 MWth [megawatts thermal]), plans to use standard PWR fuel assemblies but a somewhat shortened fuel assembly length, with an expected discharge burnup of 41–60 GWd

(gigawatt-day)/MTHM, and to discharge a spent nuclear fuel inventory of 20–29 MTHM/GWe-yr (gigawatt electrical year) (NuScale Power, 2022a). The NuScale data are comparable to a discharge burnup of 50 GWd/MTHM and spent nuclear fuel inventory of 22 MTHM/year for a 1.0-GWe PWR considered as a reference in DOE's 2014 Nuclear Fuel Cycle Evaluation and Screening (NFCE&S) report (Wigeland et al., 2014) (see Appendix E). NuScale has no plans to reprocess the spent fuel, and as a result, it would go to a deep geologic repository for disposal. Heat production from NuScale spent fuel should be similar to that of existing PWRs at an equivalent level of burnup.

iPWRs may depart from existing large PWRs in their production of GTCC waste streams. Because iPWR cores are smaller, ex-core components, including the reactor vessel, baffles, reflectors, and the steam generator, will be exposed to higher neutron fluxes than in a large PWR in which this equipment is further from the core (Brown et al., 2017). With the proper use of core baffles and reflectors, however, the core power distribution could be designed similarly to that of large PWR cores, with reduced ex-core neutron leakage, and extended fuel cycle length (Suk et al., 2021). In fact, the NuScale design certification application (NuScale Power, 2020) presents the total heat flux hot channel factor of 1.86 (compared with 2.60 in the AP1000 design control document) (Westinghouse Electric Company, 2003), indicating that the core power distribution for the small NuScale core could be flatter than that for the large AP1000 core. NuScale provided to the committee an update on its newest 250-MWth design NuScale Power Module (NPM) (NuScale, 2022b). Regarding low-level wastes (including GTCC), the update stated that the NuScale design has fewer components that can become long-lived intermediate-level waste from neutron activation compared with currently operating boiling water reactors and PWRs. At the end of life, after fuel is removed, about 95 percent of the radioactivity is associated with major components (e.g., reactor pressure vessel), which after an appropriate cool-down time (~10–15 years) are removed to support shorter decommissioning times and are classified as low-level waste. A portion of materials to be handled during decommissioning never gets contaminated or activated and can be released from nuclear control as nonradioactive waste (NuScale, 2022b). The same NuScale update mentioned that it plans to submit its standard design application amendment for the 250-MWth NPM for formal U.S. NRC review by the end of 2022 (NuScale, 2022b).

5.5.2 Waste from High-Temperature Gas-Cooled Reactors

Operation of all high-temperature gas-cooled reactors (HTGRs) will result in a unique set of waste streams compared with those of LWRs. HTGRs (and one molten salt–cooled reactor being developed by Kairos) propose to use TRISO particles as the basic building block for the fuel. See Section 4.2.4.1 for discussion of TRISO fuel development and production and Figure 4.1 for an image of a TRISO particle. Here, the focus is on the waste aspects of TRISO. The multiple pyrolytic layers making up the TRISO particles include a semiporous buffer layer that provides for long-term retention of gaseous fission products generated during reactor operation at high temperatures (Ball and Fisher, 2008), possibly in excess of 1200 K for the helium-cooled Xe-100 pebble-bed reactor design from X-energy.[7] The entire fuel assembly has radionuclides associated with fission events within the kernels and can contain uranium impurities and fission products in the matrix because of impurities or defects in the coated particles, as well as activation products from other such impurities as lithium and chlorine, and radiocarbon produced by activation of nitrogen-14, carbon-13, and oxygen-17 (Grambow et al., 2010). Typical effects observed in coated particle fuel from neutron irradiation include kernel swelling, pressure buildup, densification of the buffer layer, formation of gaps between layers, and reduction in the tensile strength of silicon carbonide. All of these phenomena may increase the probability of particle failure, although significant coating failure has not yet been observed (Grambow et al., 2010).

Although the helium coolant used in HTGRs will not become radioactive itself, it will become contaminated by radionuclides from failed fuel particles and by radioactive carbonaceous dust from the graphite moderator (Xu et al., 2020b). Radioactive carbonaceous dust production is expected to be especially high in pebble-bed reactors from friction between the pebbles as they move past each other in the reactor core, as shown in the German pebble-bed

[7] As discussed in Box 5.2, the multiple pyrolytic layers may also be able to act as primary barriers to the release of fission products during geologic disposal, although additional research is required to determine their effectiveness.

test reactor AVR, which operated from 1967 to 1988 (Humrickhouse, 2011; von Lensa et al., 2020). Estimated dust production rates range from 15 kg/yr to 100 kg/yr (Humrickhouse, 2011). Some radioactive contaminants in the helium coolant and dust will plate out or infiltrate pores on core components, making these components low-level or GTCC waste. Radioactive carbonaceous dust also presents an additional worker ingestion hazard. Furthermore, it can complicate reactor decommissioning, as it did with the AVR: the heavily contaminated core also had spent fuel pebbles stuck in crevices and cracked graphite reflectors, requiring the entire core to be filled with concrete to stabilize these materials before being transported to storage (von Lensa et al., 2020).

By far the largest volume of waste from HTGRs is graphite. Some of this graphite is in the form of neutron reflectors and other core components, which would be treated as GTCC or low-level waste. Strategies for the treatment, storage, and disposal of this type of graphite waste are discussed in Box 5.1. The vast majority of graphite waste from HTGRs, however, is from the TRISO-based fuel forms. Direct disposal of TRISO fuel (i.e., if the particles are not extracted from their graphite prisms or pebbles) would result in very large volumes of waste to manage. One estimate of waste produced from a 200-MWe pebble-bed HTGR is 90,000 spent fuel pebbles,

BOX 5.1
Graphite Waste Management

Disposal of irradiated graphite—from reflectors and moderators, and separated from fuel particles—has been a concern for several decades in countries operating nuclear reactors, in part because the amount of ^{14}C in the reflector and fuel often exceeds the licensed capacity of the disposal facility. Irradiated graphite disposal is also challenging because of its low radioactivity but very large volume and possible risk of overheating and fire. The United Kingdom is the leader in the field, as its inventory of irradiated graphite is the largest in the world. The United States has been involved in two projects with the International Atomic Energy Agency, IMMONET (a repository for data and reports on irradiated graphite) and GRAPA (GRAphite Processing Approaches), launched in recent decades to establish a knowledge base for topics related to irradiated graphite wastes. The European Commission has further developed a network of experts—via the CARBOWASTE (Carbonaceous Waste) program—to assess technologies for the characterization, retrieval, treatment, recycling, and disposal of irradiated graphite. This box describes the irradiation of graphite in reactors and considerations for storage and disposal of irradiated graphite, including potential treatment methods for decreasing its radioisotopic content.

Graphite in Reactors

Graphite has been used as a reactor moderator for a variety of reactor designs in various countries. In the United States, graphite-moderated reactors have operated at Hanford, Savannah River, Oak Ridge, Brookhaven, Pacific Northwest, and Argonne National Laboratories, and Fort St. Vrain. The graphite block configuration has evolved with reactor designs. For the proposed advanced reactors, graphite serves as both a moderator and structural material in high-temperature gas-cooled reactors and only as a moderator in molten salt reactors.

Upon exposure to neutrons, graphite becomes irradiated. Impurities in the graphite result in production of ^{3}H ($t_{1/2}$ = 12.3 y), ^{14}C ($t_{1/2}$ = 5730 y), and ^{36}Cl ($t_{1/2}$ = 300,000 y) in irradiated graphite. ^{14}C is the predominant source of activity after a short storage period. Sources of radioactivity in irradiated graphite also include fission products (^{134}Cs, ^{137}Cs, ^{90}Sr, ^{152}Eu, ^{154}Eu, ^{155}Eu, ^{144}Ce), activation products (^{57}Co, ^{60}Co, ^{54}Mn, ^{59}Ni, ^{63}Ni, ^{22}Na), and small quantities of actinides (^{238}U, ^{238}Pu, ^{239}Pu, ^{241}Am, ^{243}Am) (Ancius et al., 2005; Bushuev et al., 2002). These radionuclides can be distributed throughout the graphite matrix (Bisplinghoff et al., 2000; Forsberg and Peterson, 2015).

Removal of irradiated graphite from the reactor, which is required both during the operational lifetime and upon reactor decommissioning, presents a challenge. Graphite blocks that have been exposed to fast-neutron radiation have a weakened structure because of distortion and cracking (von Lensa et al., 2020). Consequently, the blocks must be removed intact from the reactor or removed using alternative technology

including about 17 MT of carbonaceous waste annually (Fuks et al., 2020). On an energy-normalized basis, X-energy's Xe-100 reactor would produce 160 m^3/yr of pebbles, compared with 6.8 m^3/yr of spent fuel from a large PWR (Mulder, 2021). Terrestrial Energy's graphite moderator has a 7-year lifetime and will generate 80 MT of graphite waste (LeBlanc, 2021). A 1-GW ThorCon molten salt reactor requires 780 MT of high-quality graphite, and an estimated 100 m^3 of irradiated graphite will be generated per GWe-yr (Jorgensen, 2021). Direct disposal is also being suggested for BWXT's Advanced Nuclear Reactor (BANR) and Framatome's Steam-Cooled HTGR (SC-HTGR). Moreover, the disposal volume must be further expanded to prevent overheating or fire from the release of stored energy from radiation damage. Prior research into the feasibility of direct disposal of TRISO fuel, which has mostly been performed in Europe, is summarized in Box 5.2 and discussed in more detail in Appendix G, and demonstrates that much research is still required to qualify TRISO fuel for direct geologic disposal.

Because of the very large volumes of radioactive graphite wastes that would need to be managed for direct disposal of used prismatic fuel blocks or graphite pebbles, processes are being developed to separate TRISO fuel kernels from the graphite matrix to substantially reduce the volume of waste for disposal (Grambow et al., 2006;

that makes use of the crumbled graphite. As discussed in Section 5.5.2, the use of graphite as a moderator and structural component leads to a large volume of graphite wastes.

Graphite Storage and Disposal

Upon reactor dismantlement, irradiated graphite is washed and roughly decontaminated prior to storage, disposal, or treatment. Graphite obtained from reactor dismantlement can be stored for 5–10 years, after which the volume of the graphite that has become nonradioactive is withdrawn, and the resulting radioactive portion is stored for an additional period (30–135 years). At the end of the second storage period, the radioactive portion has decayed enough to be stable or treated as low-level waste (Fuks et al., 2020). Graphite can be disposed of in near-surface disposal facilities (located at several tens of meters) or in a deep geologic repository after packaging (e.g., steel, concrete boxes). The choice of disposal facility is based on the specifications of the graphite wastes.

Graphite Treatment

Irradiated graphite may be treated to decrease the radioisotopic content of the waste or as part of disposal strategies for encapsulation and deep burial. Such treatments include bulk oxidation, pressing techniques, thermal decontamination, and chemical treatments (Wickham et al., 2017). Bulk oxidation involves partial or near-complete gasification of the graphite to CO_2, which contains $^{14}CO_2$ (Worth et al., 2021). In the case of near-complete combustion, only 1–2 percent residual waste remains, with ^{60}Co, ^{137}Cs, and ^{55}Fe (Fuks et al., 2020). The ^{14}C-doped CO_2 may be captured (e.g., by precipitation with NaOH or amine separation) or released to the atmosphere (IAEA, 1995, 2004a). Additionally, CO_2 obtained from graphite treatment can be blended with CO_2 obtained from fossil fuels to dilute the $^{14}CO_2$ concentration. Pressing techniques—including hot vacuum pressing, spark plasma sintering, and hot isostatic pressing—limit the large cavities in the irradiated graphite blocks (Lee et al., 2013). These techniques involve crushing the irradiated graphite and mixing it with a matrix (e.g., glass, bentonite) before pressing it to produce a waste form that has negligible porosity and is impermeable to water (Tan et al., 2021). Thermal decontamination of irradiated graphite decreases its radioactivity by oxidizing the graphite to generate CO and CO_2, both doped with ^{14}C (Worth et al., 2021). As with bulk oxidation, ^{14}C-doped CO_2 may be captured by precipitation at high pH (IAEA, 2004a). Chemical treatments include the high-temperature molten salt treatment, through which irradiated graphite is dissolved in acidic materials of limited solubility (Grebennikova et al., 2021). This treatment has the potential to separate actinides and fission products, hence decreasing the activity of the wastes; however, it generates additional LiCl and KCl wastes that are contaminated with actinides and fission products. Additional emerging treatments options include microwave heating and electrochemical decontamination (Bespala et al., 2019; Sivagami et al., 2019).

BOX 5.2
Direct Geologic Disposal of TRISO Fuel

TRistructural ISOtropic (TRISO) particles serve as a useful example of the type of research and development (R&D) required to qualify a fuel for direct disposal in a geologic repository. The majority of the R&D on TRISO fuels has occurred in Europe, and while there has been more research on the disposal requirements for TRISO than for other non–uranium-dioxide fuels, the work is still far from complete. As an example, although the ability of TRISO particles to retain fission products in a repository environment is often claimed as a key advantage of this fuel form, it has yet to be sufficiently demonstrated. A brief summary of prior work and remaining research needs is presented below, and a more detailed discussion can be found in Appendix G.

Studies to determine the feasibility of direct disposal of spent TRISO fuel in a geologic repository have focused on understanding the properties and performance of the graphite matrix and silicon carbide (SiC) layers surrounding the fuel kernel (Fachinger et al., 2006; Gerczak et al., 2020; Grambow et al., 2008, 2010; Katoh et al., 2012; Liu et al., 2020; Malherbe, 2013; Nabielek et al., 2009; Peterson and Dunzik-Gougar, 2011; van den Akker and Ahn, 2013). Specifically, research has been performed to examine water penetration into the graphite matrix, corrosion of the SiC coating, and potential for radionuclide release, all under a variety of physical and chemical conditions relevant to repository environments (Fachinger et al., 2006; Gerczak et al., 2020; Grambow et al., 2008, 2010; Katoh et al., 2012; Liu et al., 2020; Malherbe, 2013; Nabielek et al., 2009; Peterson and Dunzik-Gougar, 2011; van den Akker and Ahn, 2013). In general, because of the multiple barriers within the waste form and matrix that provide containment of fission product elements, TRISO fuel elements may be considered "well designed" for direct disposal in a geologic repository (Fachinger et al., 2006).

Nonetheless, additional work is needed to better understand the behavior of irradiated TRISO fuel in a repository environment. For example, a more detailed understanding is required of the effects on irradiated coating layers at different temperatures and timescales, and on radionuclide transport that occurs within the TRISO particle as a result of radiation-induced degradation of the carbide coatings (Fachinger et al., 2006; Gerczak et al., 2020; van Rooyer et al., 2018). On the disposal side, a better understanding is needed of radionuclide solubility and release, water penetration of graphite, and corrosion mechanisms of SiC and other carbide layers under various geologic conditions (Fachinger et al., 2006; Fukuda et al., 1982; Grambow et al., 2008, 2010; Malherbe, 2013; Morris and Bauer, 2005; Nabielek et al., 2009; Peterson and Dunzik-Gougar, 2011; van den Akker and Ahn, 2013).

Observation

The coated particles (kernels) of TRISO fuel have the potential advantage of providing a multibarrier waste, as the SiC layer is able to contain fission products. Demonstrating the effectiveness of TRISO kernels in retaining fission products will require carefully controlled experiments that document the distribution of fission product elements after in-reactor irradiation and identify their form and chemical species, as well as studies of the corrosion and alteration of SiC (or other carbide) ceramics as a function of type and dose of irradiation, temperature, and fluid compositions characteristic of different geologic settings. Additionally, disposal of the large volumes of radioactive graphite that will be generated remains a major issue.

Guittonneau et al., 2010). However, the removal of graphite from TRISO particles is still an immature technology. Somewhat better-established methods to remove the TRISO particles from the graphite matrix include combustion and mechanical separation, whereas experimental methods include electrochemical processing (Fuks et al., 2020). Combustion results in the carbon in the graphite being burned off in the form of radioactive carbon dioxide (containing carbon-14) (Forsberg and Peterson, 2015; Fuks et al., 2020). In past experiments with this method, the radioactive carbon dioxide was released to the atmosphere, an undesirable outcome currently because of the potential radiation doses to the public and the negative impact on climate change from the carbon dioxide. The

radioactive gases containing carbon dioxide could be captured and converted into solid calcium carbonate for disposal as GTCC or low-level waste (Forsberg and Peterson, 2015; Fuks et al., 2020), but this would add cost. Alternatively, graphite-containing spent fuel can be mechanically crushed to remove much of the graphite from the TRISO particles (Forsberg and Peterson, 2015; Fuks et al., 2020). The resulting graphite waste may contain significant radionuclides, depending on the process used, and will need to be treated as either GTCC or low-level waste. The graphite waste treatment methods described in Box 5.1 could also be applied to any graphite that is separated from the TRISO fuel particles. Additionally, any of these treatment methods would likely create additional GTCC and low-level waste streams that would need to be accounted for.

5.5.3 Waste from Sodium-Cooled Fast Reactors

Two developers of sodium-cooled fast reactors (SFRs) presented their reactor designs to the committee. ARC-100 (Advanced Reactor Concepts, LLC) proposes to use metal alloy (U/Zr) sodium-bonded fuel with 3-zone HALEU enrichment. ARC plans to treat the sodium-bonded spent fuel by pyroprocessing and to recover transuranic elements for fabrication into new fuel (Sackett and Arthur, 2021). Natrium (TerraPower) will be developed as a once-through fuel cycle in a two-phased approach (Hejzlar, 2021). In the first demonstration phase, a sodium-bonded metallic fuel will be employed (HALEU enrichment with 10 weight percent Zr [U-10Zr]) sodium-bonded to high-tensile-9 (HT9) cladding. The second phase, which would start 6–8 years after the demonstration start-up, involves the transition to a full core of an advanced metallic fuel without sodium bonding to the cladding.

While the specifics of waste treatment and disposal may vary based on reactor design, this section focuses on waste issues common to all SFRs: (1) radioactive sodium waste streams during the operation and decommissioning phases and (2) management of the spent nuclear fuel/high-level waste (SNF/HLW) for both sodium-bonded spent fuel and non–sodium-bonded spent metallic fuel.

5.5.3.1 Unique Waste Streams from Sodium-Cooled Fast Reactors During Operation and Decommissioning

Multiple waste streams are expected from operating and decommissioning an SFR. There will be routine operational waste, including solid, liquid, and gaseous waste, for which technology has been developed. All fast-spectrum reactors, including lead-cooled reactors (e.g., LeadCold's SEALER-55 and Westinghouse's LFR) and molten salt reactors (e.g., TerraPower's MCFR and Moltex's Stable Salt Reactor) use reflectors and shielding, which become irradiated and contaminated and will require disposal as GTCC in the United States or intermediate-level waste for operation outside the United States. In addition to the low-level liquid and solid waste generated during normal operations, TerraPower suggests that its Natrium SFR will produce about 80 MT of reflectors and other reactor-vessel irradiated internals as low-level (Class C) waste, 70 MT of irradiated control rods also as low-level Class C waste, 4–8 m^3 of activated primary cold traps (containing cesium and tritium) that will require storage with decay and treatment before disposal as GTCC (TerraPower, 2021b). The ARC-100 Advanced SMR will produce 67 MT of reflectors and shielding and 4.6 MT of control assemblies, as well as GTCC and low-level waste (Sackett and Arthur, 2021).

The most unique and challenging waste streams generated from SFRs are those associated with the liquid sodium coolant (IAEA, 2019e). Volumes of sodium coolant will vary depending on the reactor design (e.g., pool type versus loop type) and reactor power. As an example, TerraPower indicates that its Natrium reactor will produce 800 m^3 of activated sodium (Hejzlar, 2021). Part of the challenge of dealing with this material is that sodium reacts with moisture and air to pose a significant fire hazard. It should also be noted that sodium wastes are Resource Conservation and Recovery Act wastes subject to state regulation. Although these challenges are unique to SFRs, they are not new, and there is some experience from decommissioning of several sodium-cooled reactors. Special processes have been developed to manage activated sodium and sodium-contaminated materials (IAEA, 2007b). Sodium coolant from decommissioned SFRs may be used in new reactors (IAEA, 2007b).

A key difference between sodium waste treatment during operations and decommissioning is the waste volume that must be handled. The waste quantities are considerably larger during decommissioning because all sodium coolant will finally be drained and become waste. For example, upon decommissioning the Experimental Breeder Reactor II (EBR-II) produced a total of 327 MT of sodium coolant, and the Fast Flux Test Facility

(FFTF) generated 900 MT of sodium coolant (IAEA, 2007b). The bulk sodium coolant contains several sources of radioactivity, including sodium activation products and contamination from actinides and fission products. The main radionuclides include sodium-22, tritium, cobalt-60, and cesium-137.

During decommissioning of an SFR, in addition to the large quantity of sodium coolant that needs to be drained, a significant amount of residual sodium usually remains. For example, residual sodium in EBR-II amounted to about 1300 kg in the primary vessel and 400 kg in the secondary circuit (IAEA, 2007b). There are several sources of "residual sodium on and within components removed from sodium circuits. Residual sodium may consist of a thin layer wetting the surfaces or may be larger quantities retained in pockets that cannot be drained during component removal" (IAEA, 2007b). The residual sodium needs to be removed because of the high chemical reactivity of sodium. Unloaded fuel assemblies would also have to be cleaned of residual sodium before storage in air or water. Additionally, "any maintenance, repair, or modification carried out on a circuit or a component wetted with sodium will generate sodium-contaminated gloves, tools, etc." (IAEA, 2007b). Importantly, while storing such sodium-contaminated items, it is necessary to avoid contact with air due to the potential fire hazard.

Many sodium treatment technologies have been developed over the years, in both the nuclear and industrial sectors, where sodium metal is used in many processes and applications. Some techniques involve pretreatment to remove cesium and tritium, in order to reduce the waste classification for disposal. The majority of treatment processes involve phases of neutralization and separation of liquid sodium from the metal surface. There are two proven technologies for the treatment of bulk sodium: (1) the NOAH process, involving a highly controlled reaction of sodium with water to produce liquid sodium hydroxide that is subsequently neutralized, and (2) the Argonne process used for EBR-II and Fermi-1, where a caustic process is used to react sodium with aqueous sodium hydroxide solutions to generate a concrete-like sodium hydroxide monohydrate crystal that can be packed in drums (IAEA, 2007b).

A variety of technologies are being considered for conditioning the sodium waste for storage and disposal, including drying solidification, cementation, conversion into dry carbonates, and conversion into dry sodium salt. Although many of these technologies are at a high level of technical readiness, few are currently ready to be used for commercial-scale applications.

5.5.3.2 Spent Nuclear Fuel and High-Level Waste

Most fast reactors, including SFRs, use metallic fuels, which consist of a heavy metal (typically U) that is alloyed with various other metals. As described in Chapter 4, metallic fuel is usually bonded to cladding with liquid sodium to improve heat transfer. Typical fuel materials include U-Zr and U-Pu-Zr, and cladding materials are austenitic stainless steels or ferritic-martensitic steels. TerraPower has proposed a non-sodium-bonded metallic fuel comprised of U metal annular fuel with He-filled central pore and advanced ferritic-martensitic steel cladding to reduce void swelling.

Traditionally, sodium-bonded spent nuclear fuel falls into two categories: driver fuel and blanket fuel. After irradiation of a driver fuel, some of the metallic sodium enters the metallic fuel and becomes inseparable from it. In addition, spent driver fuel and cladding components interdiffuse to such an extent that removal of sodium is not practical by mechanical stripping of the fuel cladding. On the other hand, irradiation of blanket fuel results in little metallic sodium diffusion into the fuel, thus allowing mechanical stripping of the spent blanket fuel cladding (DOE, 2000).

The Natrium reactor concept proposed by TerraPower includes the development of an advanced metallic fuel (Type 1B fuel) that does not use sodium to bond the fuel to the cladding. Instead, Type 1B fuel uses annular fuel slugs with an inert gas–filled central hole, with the fuel having direct thermal contact with an advanced ferritic-martensitic steel cladding material with very low void swelling (Hejzlar, 2021).

Preparing for direct disposal of the Natrium Demo spent fuel to a geologic repository will require significant effort, including gathering the necessary technical information and verifications, because the chemical composition, radionuclide composition, and material properties of the Natrium Demo spent fuel (both Types 1 and 1B) are different from U.S. LWR spent fuel.

5.5.3.3 Treatment of Sodium-Bonded Spent Metallic Fuel

Due to the exothermic reaction of sodium with water, spent sodium-bonded metallic fuel has not been considered acceptable for direct disposal in any geologic repository and therefore must be processed prior to disposal to develop acceptable waste forms. DOE has been conducting research and development (R&D) to treat driver and blanket spent sodium-bonded metallic fuel from EBR-II using electrometallurgical technology (EMT) or pyroprocessing. As discussed in Chapter 4, pyroprocessing consists of electrorefining the spent fuel in an electrochemical cell and allows for the separation of the fuel into (1) metallic uranium, (2) a metallic waste form, and (3) a highly radioactive salt mixture (National Research Council, 2000). In DOE's 2000 Record of Decision (DOE, 2000) choosing EMT for the treatment of sodium-bonded spent metallic fuels, it also assessed other treatment options such as plutonium uranium reduction extraction (PUREX) and "melt and dilute." More recently, DOE is considering a "melt, distill, and dilute" process to deal with the sodium-bonded spent fuel from the Versatile Test Reactor planned for Idaho National Laboratory (Crawford, 2020). The resulting waste streams would include highly radioactive ingots, contaminated sodium hydroxide, and the condensable fission products, as well as nonrecyclable fuel production scrap (Crawford, 2020).

5.5.3.4 Wastes from Electrometallurgical Technology (Pyroprocessing)

As detailed in Sections 4.3.6.3 and 4.3.6.5, a number of waste streams are generated from pyroprocessing. The chopped spent metallic fuels are placed in an anode metal basket and immersed in a 500°C molten LiCl and KCl salt. When current is passed through the metal baskets, fission products and actinides are oxidized and dissolved in the salt bath. The U is reduced to its metallic form and accumulates on the cathode. Cladding and noble metal fission products remain in the anode and can be cast into metal ingots and become metal high-level waste forms. Fission products in the salt bath are first passed through zeolite columns, then mixed with glass and pressed into a glass-bonded sodalite, a ceramic form of high-level waste. Theoretically, U and actinides in the cathode can be recycled.

5.5.3.5 Waste Forms for Sodium-Bonded Spent Metallic Fuel

Because of the high sodium concentration, salt waste cannot be processed into glass waste form. Instead, it is blended with zeolite and glass into a glass-bonded sodalite composite ceramic high-level waste form (Hall et al., 2019a). This ceramic waste form can immobilize radionuclides in the ceramic structure and dilute the actinide concentrations. Furthermore, the chloride salts are incorporated into the cages of the zeolite structure, forming salt-loaded sodalite, which makes the salt much less soluble and less corrosive when in contact with water (Hall et al., 2019a). The EMT salt waste will generate a large mass of ceramic waste form—it is estimated that 1.72 MT of EBR-II salt waste will produce 50.95 MT of ceramic waste form (Frank and Paterson, 2014).

The predominant composition (about 90 percent) of the metal high-level waste form is irradiated stainless-steel cladding. The metal high-level waste form has two main phases interspersed in the microscopic scale: an intermetallic Zr (Fe, Cr, Ni) phase and an Fe solid solution phase. ^{234}U and ^{99}Tc are the primary dose contributors, and the amount of heat-generating radionuclides is very small, so the thermal output of the metallic high-level waste form is negligible.

DOE research has concluded that ceramic and metallic waste forms converted from spent metallic fuel are acceptable for repository disposal (Hall et al., 2019a). For these waste forms, radionuclide release rates would depend largely on repository and fuel conditions, which include temperature, fluid chemistry, and radiation level. Rechard et al. (2017) also proposed direct disposal options for the salt waste: (1) deep borehole in crystalline basement rock and (2) the Waste Isolation Pilot Plant (WIPP) in bedded salt in New Mexico. In the deep borehole disposal concept, the salt waste would be placed in containers at the bottom 1–2 km of an approximately 5-km-deep borehole. According to Rechard et al. (2017), the advantages of the deep borehole disposal concept include (1) restricted migration of radionuclides due to low permeability in deep crystalline rocks, (2) little interaction between deep and shallow fluids, (3) limited solubility and enhanced sorption of radionuclides in a deep reducing

environment, and (4) further restricted mobility of radionuclides by the high salt content of the waste. The study performed thermal-hydrologic analysis of the disposal concept, demonstrating its feasibility. Rechard et al. (2017) also described a roadmap for the disposition of salt waste as remote-handled transuranic waste at WIPP, which may be technically feasible but would encounter many regulatory and legal obstacles, such as modification of the WIPP permit and an environmental impact assessment.

A 2019 assessment performed by the Center for Nuclear Waste Regulatory Analysis, prepared for the U.S. NRC, found:

> Conversion of spent metal fuel to ceramic waste immobilizes radionuclides in the ceramic structure and dilutes Pu concentration, but this ceramic waste form is subject to dissolution, as salt-containing ceramic materials are hygroscopic, as well as to radiation embrittlement. Metallic waste could be subject to oxidation, pitting, and galvanic corrosion, and when chloride is present in the disposal environment, localized corrosion would be enhanced. For the spent blanket fuel, the presence of water inside the disposal package can cause uranium to react and form uranium oxides and hydrogen gas, some of which can be absorbed into the spent metal fuel to form uranium hydrides, which are pyrophoric materials that can challenge waste package performance. (Hall et al., 2019a)

For these waste forms, radionuclide release rates would depend largely on repository and fuel conditions, which include temperature, fluid chemistry, and radiation level.

5.5.4 Waste from Molten Salt Reactors

Molten salt reactors (MSRs) come in a variety of design concepts and are adaptable to a wide range of fuel cycles, as discussed in Sections 3.2.3.5 and 4.3.6.5. As a result of these different designs, the specific waste streams will vary, but there are general similarities across all MSR design concepts. MSR waste streams can include off-gases, fuel-related waste streams, graphite or carbon components, metallic reactor components, and operational and decommissioning wastes. Although many of the MSR wastes are different from LWR-type wastes, researchers have identified potential pathways for management and disposition of each potential MSR waste (Riley et al., 2018, 2019). However, spent fuel storage on-site at MSRs will differ significantly from that of LWRs. Because of the air- and water-sensitivity of the salt, it may require new interventions to stabilize it for storage or before transfer to a waste repository (Riley et al., 2018).

Two waste streams will carry most of the radionuclides generated during MSR operation: off-gas wastes and the fuel salt itself. Because MSR fuel is liquid, the volatile radionuclides are not contained by fuel cladding, and as a result, the noble gas fission products (such as Xe and Kr, which are also precursors to decay products such as Cs, Ba, Rb, and Sr), as well as aerosolized salts, particulates, reactive gases (I_2, Cl_2, F_2, HF), halides, O_2, N_2, and ^3H must be captured by an off-gas system (McFarlane et al., 2020; Riley et al., 2019). Components of the off-gas system vary depending on the design but can comprise decay tanks and piping and filter banks to confine the radionuclides as they decay. Each of these components needs to be maintained during operation and treated separately for final disposal depending on the nature of the radioactive species trapped within the unit (McFarlane, 2021; Riley et al., 2019). The particulates, aerosols, and reactive gases (I_2, Cl_2, F_2) can be immobilized in a ceramic waste form; the ^3H waste is more difficult to treat, as are the residual halides (McFarlane et al., 2020; Riley et al., 2019). The noble gases, such as Xe and Kr, can be stored in the off-gas system and allowed to decay (Riley et al., 2019). Development of an off-gas system continues to be an integral part of reactor design. Research into both systems and individual components is ongoing. For instance, a molten hydroxide scrubber is being developed for aerosol and acidic gas removal, and metal organic frameworks have been developed for noble gas separations. Both are being studied for MSR applications (Riley et al., 2019).

Carbon-related waste steams will be generated in some MSR designs where graphite is used as a moderator or reflector. Neutron radiation damage affects the lifetime of the carbon-based components, and for many of the reactor designs may result in the need to replace the graphite moderator approximately every 7 years, depending on such factors as reactor flux and salt type (Riley et al., 2019). These graphite components will experience both surface contamination and diffusion of radionuclide species into graphite pore space (Forsberg et al., 2017; Riley

et al., 2019) (see Box 5.1). Thus, the relatively large volume of carbon-based waste streams might pose disposal challenges unless they can be sufficiently decontaminated to low-level waste (Riley et al., 2019). To reduce the amount of high-level waste, these materials may first require processing (instead of direct disposal) to remove fission products, salt, and ^{3}H, followed by compaction prior to disposal. It is also possible that the graphite could be recycled once most of the radionuclides have been removed, although this is unlikely because of residual ^{14}C (Riley et al., 2019).

In addition to graphite reactor components, metal reactor components will become activated and require disposal as either GTCC or low-level waste, which could be an option for other wastes from decommissioning and operating streams if they can be reduced in size and/or decontaminated (Riley et al., 2019). Salts and the reactive gases (e.g., I_2, HF, Cl_2) are very corrosive to metallic materials, such as those found in the reactor vessel, although their effects will be minimized by redox control. Resulting corrosion products, such as Cr and Fe, will tend to transport from the hot zone to the cooler areas in the reactor primary circuit, where they are expected to plate out on the surfaces. Metal surfaces after decommissioning will be coated with salt and insoluble noble metal fission products (e.g., Mo, Pd, Rh, Tu, Tc, Nb, Sb, Ag) that tend to plate out on reactor surfaces (Riley et al., 2019). These materials may either be decontaminated and recycled or disposed of directly in a repository. While decontamination processes will result in additional waste streams, this is not unique to MSR wastes. Such an effort will require a cost-benefit analysis.

The MSR fuel salts have high water solubility, unlike ceramic wastes generated from LWRs. Thus, managing and disposing of the fuel salt itself and containment of the radionuclides poses a challenge. One option would be to simply allow the fuel salt to solidify and dispose of it directly in a deep geologic repository, especially in bedded salt or a salt dome. However, water movement through a salt repository or pressurized brines will transport radionuclides. Geologic repository environments other than salt would be inappropriate for direct disposal because of the potential for water to dissolve the fuel salt waste, readily releasing radionuclides into the subsurface. Even in salt-based repositories, concerns about interaction with a brine producing either a soluble U phase from the hydration of crystalline UF_4 or the production of hydrofluoric acid from UF_4 would exist (Grenthe et al., 1992; Kozak et al., 1992; Tracy et al., 2016). As a result, most analysts considering fuel salt disposal are planning to first process the salt into more durable waste forms (McFarlane, 2021; Riley et al., 2019; Terrestrial Energy, 2021).

Thus, salt-based waste must either be dehalogenated or the halide forms must be stabilized before disposal. Treating MSR fuel salts to produce waste forms acceptable for disposal in a geologic repository will have to be capable of containing all the different species present in the salts, including halides, alkalis, alkaline earths, rare earths, and actinides, although it is likely that the actinides will be removed before disposal. As a result, a combination of possible waste forms, including ceramics or mineral waste forms, ceramic metals (cermet), halide metals (halmet), and possibly glass waste forms are being considered (Riley et al., 2018, 2019). Numerous separation technologies can be used to process the salt, including reductive extraction, oxidative precipitation, distillation, melt crystallization, dehalogenation, phosphorylation, ion exchange, and a glass material oxidation and dissolution system (Riley et al., 2019). However, as noted by Riley et al. (2019), no single technology will provide the solution, but "management strategies exist for all types of waste based on demonstrated technologies. These management strategies range from simple to complex and from direct disposal to separating, partitioning, and recycling fractions of the waste for future MSRs."[8]

Immobilizing MSR fuel salts may have a significant impact on the volume of high-level waste. Previous experience with production of ceramic waste forms from the salt wastes resulting from the pyroprocessing of EBR-II spent fuel resulted in a 30-fold increase in waste mass, with 1.72 MT of EBR-II waste producing 50.95 MT of ceramic waste form (Rechard et al., 2017). In addition, all of these salt waste processing technologies will generate additional GTCC and low-level waste streams, which will have to be accounted for and disposed of. Conversely, unlike pyroprocessing, additional salt does not need to be added to solubilize the spent nuclear MSR fuel. As most carrier salts for both fluoride and chloride reactors require isotopic enrichment, there are schemes for recycling the carrier salts. Finally, the MSR technology is being considered for "waste burning," in order to

[8] One committee member does not agree that management strategies have been demonstrated for all types of wastes from MSRs at this point in time. If that were the case, the Oak Ridge Molten Salt Reactor Experiment wastes would already have been dealt with; they have not.

reduce the inventory of actinides that will comprise high-level waste. These factors must all be considered in the determination of the masses, volumes, or compositions of these additional waste streams.

As there is no operating repository, the fuel salt will need to be stored at the reactor site prior to disposal. Experience from the Molten Salt Reactor Experiment shows that solid or frozen fuel salt in storage tanks can generate fluorine and volatile U species via radiolysis (Haghighi et al., 2002; McMillan, 2019). This situation arose because of residual actinides remaining in the salt even after most of the U had been removed by fluoride volatility. A 1997 study assessed that "criticality becomes a concern when more than 5 kg of uranium concentrates to over 8 weight percent of the salt in a favorable geometry" and advised avoiding that situation (Hollenbach and Hopper, 1997).[9] To mitigate the accumulation of hazardous UF_6 gas, some designers plan to heat spent fuel in storage tanks to more than 200°C to increase the recombination rate of the products of radiolysis (Terrestrial Energy, 2021). Until the salt is processed into a stable waste form, storage tanks will need monitored off-gas collection systems to mitigate this risk.

Additional waste streams, common to all reactor designs, will include operational wastes from equipment, personal protective equipment, high-efficiency particle air filters, and other operating materials (Riley et al., 2019; Terrestrial Energy, 2021), as well as waste streams dominated by GTCC and low-level wastes from the decommissioning of these reactor facilities.

5.5.5 Characteristics of Spent Nuclear Fuel and High-Level Waste from Advanced Reactors and Their Impacts on Geologic Disposal

As noted in the earlier discussion, the spent nuclear fuel isotopic compositions and evolution as a function of time is important information for geologic disposal. Other than for the Natrium SFR from TerraPower, there is little detailed information provided by most advanced reactor vendors on the characteristics of spent nuclear fuel isotopic compositions. The committee therefore decided to use the information from DOE's 2014 *Nuclear Fuel Cycle Evaluation and Screening* (*NFCE&S*) report (Wigeland et al., 2014) to supplement the assessment of potential impacts of advanced reactors on geologic repository disposal. Although the information from *NFCE&S* may not exactly represent the advanced reactors being developed by the vendors, it does capture key features of the designs that are similar to some of the advanced reactors. Three metrics used by *NFCE&S* in its assessment of waste management impacts were selected by the study committee: "(1) mass of SNF/HLW disposed per energy generated, (2) activity of SNF/HLW at 100 years per energy generated, and (3) activity of SNF/HLW at 100,000 years per energy generated" (Wigeland et al., 2014). In particular, the committee selected data from *NFCE&S* for three reactor designs: the high-temperature gas-cooled reactor (HTGR), the molten salt reactor (MSR), and the pressurized water reactor (PWR) (as the base case for comparison). The corresponding data for the TerraPower SFR are also compared. A two-tier coupled SFR-PWR system is also analyzed in Appendix E to indicate the potential impact of recycling the Pu generated in both the PWR and SFR cores. The symbiotic fuel cycle illustrates the synergistic arrangement where the Pu produced, together with the remaining U, is recycled, and minor actinides and fission products are sent to a repository. The only feed material required is a small amount of natural U for a combined system replenishing the fission products and minor actinides to be sent to a geologic repository. Appendix E provides details of data and information from TerraPower and *NFCE&S* and shows the following results for each type of advanced reactor (see Tables E.3–E.5 in Appendix E):

SNF/HLW mass per energy generated: For the three advanced reactor designs, reduction in the masses of SNF/HLW requiring geologic disposal per unit of electricity generation compared with the base case PWR ranges from a factor of 3 to about one order of magnitude. These results are to be expected for unit energy generation,

[9] The Molten Salt Reactor Experiment Remediation Project was initiated by Oak Ridge National Laboratory to stabilize conditions in the facility by treating and removing UF_6 from fuel and flush salts, and converting it to the stable oxide form for storage, pending ultimate disposal. As of 1998, "All significant quantities of fissile material are under appropriate criticality safety controls. No portion of the off-gas system is at a positive pressure with respect to its surroundings. The potential for an energetic chemical reaction in the charcoal bed has been eliminated" (Peretz et al., 1998). Currently, surveillance and maintenance at this facility is ongoing, and decommissioning concepts are being evaluated (McMillan, 2019).

since higher enrichment, higher burnup, and any recycling will reduce the SNF/HLW mass. This mass reduction indicates the somewhat better utilization of resources for advanced reactors. However, the reduced mass does not necessarily translate into advantages to disposal in a geologic repository, because the heat load of the waste (most significant) and volume of the waste (less significant) will typically impact the repository capacity management.

SNF/HLW radioactivity per energy generated at 100 years after discharge: With the fission process releasing two fission products and approximately 200 MeV (mega electron volts) of energy per fission, a 1.0-GWe (gigawatt electrical) power plant consumes approximately 1.0 Mg (megagrams)/year of nuclear fuel and produces 1.0 Mg/year of fission products, making the fission product inventory in spent nuclear fuel proportional to the thermal energy generated. Therefore, on a per-unit-energy-generation basis, the four reactor designs produce similar radioactivity from fission products, and fission product inventories in the spent fuel are reduced for designs with a higher thermal efficiency.

The contributions from short-lived fission products (e.g., ^{90}Sr, ^{137}Cs) dominate the radioactivity of SNF/HLW at 100 years and produce significant heat. Advanced reactors will in general produce higher amounts of heat in each spent nuclear fuel package because of the higher burnup resulting in the higher heat load. The thermal load of waste, in addition to the mass, volume, and other factors,[10] will impact the spacing of waste packages, repository footprints, and engineering designs.

SNF/HLW radioactivity per energy generated at 100,000 years after discharge: On a per-unit-energy-generation basis, the three advanced reactor designs, as well as the two-tier coupled PWR-SFR system, can reduce the amount of long-term radioactivity from a few percent to a factor of 2 as compared with the PWR system. These reductions, however, do not have significant impacts on the performance of the repository if geochemical and geologic conditions of the repository are carefully chosen to limit the mobility and accessibility of the radionuclides.

From the analysis summarized here and detailed in Appendix E, plus the discussion in Section 5.4.1 on the important factors for evaluating the safety of geologic repositories, the committee emphasizes the observation that *the current advanced reactor technologies do not appear to offer significant beneficial impacts to the disposal of SNF/HLW in a geologic repository*.

5.6 POTENTIAL IMPACTS OF ADVANCED NUCLEAR FUEL CYCLE WASTES ON STORAGE AND TRANSPORTATION OPERATIONS

5.6.1 Current Status

Storage and transportation management of radioactive wastes from LWR power generation has evolved significantly over the past 50 years of operations. The regulatory environment is stable, and the technology is mature. Operational experience further points to an understanding of how best to safely manage a hazardous material within the context of strict regulatory requirements. This is not to say, however, that there are no technical or regulatory issues left to be addressed. Higher burnups, increased enrichments, new fuel forms, extended dry storage, and potential storage-transportation-storage operational scenarios generate new issues not necessarily envisioned during the initial licensing of current storage and transportation systems. Proposed new fuel designs stemming from the advanced reactor initiatives will introduce issues that will need to be addressed from both regulatory and technical perspectives. Work continues on both the licensing and R&D fronts to address issues as they are identified to assure safety and regulatory compliance through the entire fuel cycle.

Given this backdrop, it is important to provide a brief background of the LWR spent fuel regulatory and technical status, as it will provide a benchmark by which storage and transportation management of advanced fuel cycle wastes will be evaluated. The Code of Federal Regulations (CFR) Title 10, Parts 71 and 72 (10 CFR

[10] The sentence was modified following a prepublication version of the report to indicate factors other than thermal load that impact the spacing of waste packages, repository footprints, and engineering designs.

71, 72) provide the primary regulatory criteria for the design of transportation and storage systems for high-level radioactive and fissile materials.

For transportation, 10 CFR 71 safety criteria address fuel containment, criticality, shielding, and thermal management. These criteria include the 9-m drop test and 30-minute fire-loading requirements for assessing a packaging's ability to maintain containment during extreme mechanical and thermal loading conditions. Part 71 also includes a nonmechanistic criticality criterion that assumes breach of containment, full moderation, and fuel spatial reconfiguration to afford the optimal potential of creating a possible criticality event. Shielding and cask surface temperature thresholds are also defined 10 CFR 71 criteria. To support the licensing process, the U.S. NRC has recently published a review guide, NUREG-2216, to assist both the licensee and U.S. NRC review team with a consistent approach for assessing the safety of a transportation packaging design against the regulatory criteria (U.S. NRC, 2020d).

For storage, 10 CFR 72 criteria provide design requirements for both wet pool and dry storage operations not associated with Part 50 (10 CFR 50), which covers pool storage requirements necessary for removal of fuel immediately from the reactor. 10 CFR 72 safety criteria address fuel confinement, shielding, criticality, and fuel retrieval, and provide guidance to environmental loadings such as earthquakes, tornadoes, floods, and temperature extremes that the storage system may be subjected to over its lifetime. Similar to the transportation regulations, the U.S. NRC has recently published a technical review guide for the design of dry storage systems, NUREG-2215 (U.S. NRC, 2020e).

Safety assessments for both storage and transportation, in principle, focus on fuel containment and confinement, criticality, shielding, and thermal management. While most licensing applications historically have been associated with LWR spent fuel, the limited body of licenses for advanced fuel cycle materials suggests that future license applications will be judged against these same regulatory criteria (DOE, n.d.-a; Hall et al., 2019b, 2020). Therefore, for the advanced fuel cycle materials addressed in this study, assessment of storage and transportation systems envisioned for these materials will be based on the current U.S. NRC regulations.

As a policy in the United States, the once-through fuel cycle depends on a geologic/engineered repository for final disposition of spent nuclear fuel. The Yucca Mountain project was the planned geologic disposal site; however, as noted above, development of this site has not been pursued, and the future of repository planning is uncertain. This has created a spent fuel management issue that looms large for the country. Figure 5.4 illustrates current (as of December 2019) and projected quantities of LWR spent fuel in storage (Freeze et al., 2021). The red line in Figure 5.4 shows the entire U.S. inventory of spent fuel as it is off-loaded from reactor operations over time and indicates that the spent fuel inventory has already exceeded the statutory capacity of the Yucca Mountain repository (70,000 MTHM, per the NWPA). The total sum of fuel in storage is divided between pool and dry storage. As the figure shows, current storage is split fairly evenly between wet and dry storage, with a continuing increase in dry storage to the point where virtually all spent fuel will be in dry storage systems. The Commercial Spent Fuel Projections (CSFP—solid lines) and Next Generation Systems Analysis Model (NGSAM—dashed lines) projections represent two different models with different input assumptions. While there are some differences in the plots, the general trends and results are similar.

Utilities have moved to dry storage systems because they ran out of space in spent fuel pools and because the capital and operational costs are well understood and predictable. Operations and maintenance are also straightforward and predictable. Passive cooling means that there are no pumps or coolants that need to be maintained, and thermal management using natural convection has been shown to be effective (Fort et al., 2019).

There are currently just over 3,200 loaded canisters in dry cask storage systems in the United States. Incremental design changes over the years have resulted in large canistered systems with payloads of up to 37 PWR assemblies or 89 boiling water reactor assemblies, loaded weights of up to 50 MT, and heat generation of up to 40 kW (Freeze et al., 2021). These canistered systems are designed for transportation as well as storage; however, they are licensed for transport at much a lower heat generation than during storage. Currently, the utilities do not intend to repackage the spent fuel for transportation after storage.

While these dry systems meet all the safety criteria for storage, downstream issues may arise from their canistered, high-payload configuration. High heat loads may require longer-term storage prior to transportation to meet the cask surface temperature limit of 50°C. Some of the licensed storage systems do not, as yet, have compatible licensed

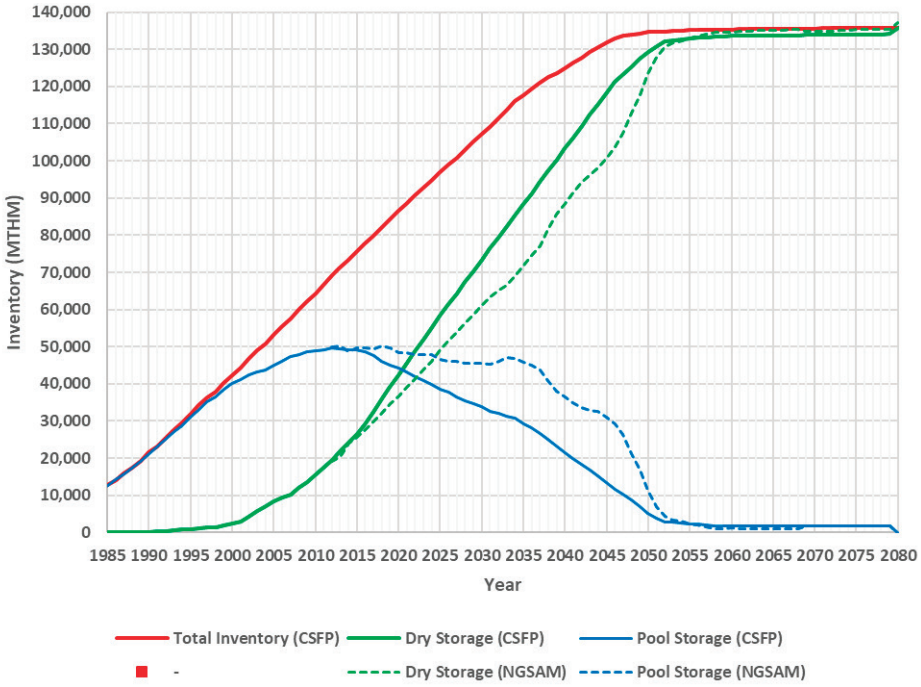


FIGURE 5.4 Current and projected spent fuel inventories in the United States.
NOTE: CSFP = Commercial Spent Fuel Projections; MTHM = metric tons of heavy metal; NGSAM = Next Generation Systems Analysis Model.
SOURCE: Freeze et al. (2021). Courtesy of Sandia National Laboratories.

transportation casks. Work is ongoing to design, test, and deploy a railcar to the American Association of Railroad's S-2043 standard. In January 2022, DOE issued a request for proposals for "the fabrication and testing of a prototype eight-axle railcar" that meets this standard (DOE, 2022b). This railcar will be needed once transport campaigns begin. The sheer size of these canisters may create operational issues when moving from storage to transportation operations. Issues such as crane capacity, facility clearances, and rail access at shutdown sites need to be addressed. The quantity of canistered systems may render repackaging unrealistic. The current inventory of 3,200 canisters is expected to top out at approximately 10,000. Costs to repackage spent fuel from existing canisters to a disposal canister have been estimated to be ~$20 billion in 2018 dollars (Freeze et al., 2021). In addition to the operational and material costs associated with repackaging, consideration must be given to the fact that the United States currently does not have the infrastructure necessary to conduct these operations at scale. Repackaging would require a significant capital investment and an accountancy for the time needed to design, permit, and build the infrastructure.

As a result, while the United States has licensed storage systems for spent fuel that meet all current storage regulatory criteria, these systems may create issues for downstream extended storage and transportation and may not be suitable for disposal. The current lack of a repository site, coupled with the lack of established design criteria for the waste packaging, creates a dilemma for how spent fuel is currently packaged for storage. It is worth noting that DOE's Office of Civilian Radioactive Waste Management (OCRWM) was established as part of the NWPA to develop a repository, as well as the infrastructure required to move spent fuel from the reactor sites to the repository. While OCRWM was engaged with the development of the Yucca Mountain site, it also had an active program to develop the transportation infrastructure to deliver commercial spent fuel to the repository. With the cessation of licensing activities for the Yucca Mountain repository, OCRWM was disbanded, resulting in an organizational lack of attention to linking important storage and transportation infrastructure issues to eventual disposal requirements. Nonetheless, in recent years, DOE's Office of Integrated Waste Management has been examining integrated approaches to storage, transportation, and disposal of spent nuclear fuel and other high-level radioactive wastes (Nutt, 2021).

Regulatory accommodation for current LWR spent nuclear fuel has been addressed to allow for long-term storage through the Continued Storage Rule (U.S. NRC, 2014a). This rule provides a pathway for continued storage of spent nuclear fuel based on three storage scenarios analyzed in the U.S. NRC's associated General Environmental Impact Statement (GEIS) (U.S. NRC, 2014b): a repository becomes available 60 years after the licensed life of a reactor (short-term storage), a repository becomes available 100 years after the short-term storage scenario (long-term storage), and a repository never becomes available. It is important to note that the U.S. NRC GEIS specifically excludes the evaluation of advanced reactors, GTCC waste,[11] high-level waste from reprocessing, and associated wastes from reprocessing. This points to a regulatory gap for long-term storage of those wastes excluded in the U.S. NRC GEIS. Arguably, this issue will not arise until decades into the future. However, it is important to recognize these gaps early and to understand the potential impacts on costs, schedules, and operations as the issues are addressed.

The lesson for advanced fuel cycle materials is this: while plans for packaging advanced fuels and materials for storage may meet regulatory criteria, their waste forms and packaging may not be compatible with downstream extended storage, transportation, and disposal regulations and needs. Developers of advanced reactors and fuel cycles need to be transparent and clear about development of storage systems for new materials and how their designs may impact downstream operations. This applies not only to spent fuel, but also to GTCC and low-level waste, as well as hazardous materials that are generated from reactor operations. As part of its Extended Storage Collaboration Program within its Used Fuel and High-Level Waste Management program, EPRI has established a task force in collaboration with industry to evaluate potential issues related to the back end of the fuel cycle for advanced reactors (EPRI, 2021b, 2022).

5.6.2 Advanced Reactor Fuels and Materials

Storage and transportation assessments for advanced fuel cycle materials may best be addressed by categorizing the fuel proposed for advanced reactors. However, given its unique impact on the advanced fuel cycles being proposed, HALEU used as a fissile feedstock, through to the back end as a component of spent fuel, will be discussed first, followed by advanced fuels.

5.6.2.1 HALEU-Based Fuels

HALEU is being proposed as source fissile feedstock for most of the advanced fuels that will feed the advanced reactors. From the presentations given to the committee regarding HALEU, there is a general concern with regard to the availability of supply in the time frames needed to support deployment of demonstration and prototype advanced reactors (see Section 4.2.3). The most likely near-term source of HALEU to supply DOE initiatives is Russia. Current world events have exposed the vulnerability of relying on nondomestic sources.

Beyond this general concern, transportation details associated with HALEU supply need to be considered with regard to schedules, cost, and regulatory gaps. For example, the International Atomic Energy Agency (IAEA) recently published an issues assessment of LWR fuel enriched to >5 percent (IAEA, 2020d). One cited Russian study in the IAEA publication analyzed the current 30B transport container for shipment of enriched UF_6. Results of the analyses show that the 30B would be feasible for UF_6 transport up to 7 percent–enriched UF_6. For enrichments greater than 7 percent, additional work would be needed to build a safety case for transport using the 30B, or the packaging may need redesigning to accommodate more highly enriched UF_6. Depending on the supply chain needed to acquire HALEU, if the enrichment and fuel fabrication facilities are not collocated, the enriched UF_6 will have to be transported to the fuel fabrication facility, thereby potentially impacting the ability to ship UF_6 in currently licensed containers. Orano has submitted a license application to the U.S. NRC for a modified 30B container, designated the DN30-X, to ship UF_6 enriched up to 10 or 20 percent. The capacity of this container will be approximately 50 percent of the currently licensed 30B (Redmond and Ashkeboussi, 2022). An alternative to gas shipment is reconversion of the UF_6 gas to an oxide or metallic form for shipment to the fuel fabrication facility. Discussion with DOE indicates that this would probably be the plan for augmenting their HALEU supply

[11] See Appendix D for the definition of GTCC waste.

(Griffith, 2021; Regalbuto, 2021). There are currently three packages licensed for feedstock HALEU in the oxide or metallic form; however, the payloads of these packages are very limited, considering the quantities of HALEU feedstock that may need to be shipped (Redmond and Ashkeboussi, 2022). The supply chain aspect of acquiring sufficient HALEU to meet DOE's needs in the advanced reactor programs requires assessment with respect to where the HALEU is sourced to determine potential impacts to program schedules.

From a domestic supply perspective, if shipment of enriched UF_6 in the 30B is required, current regulatory requirements (10 CFR 71.55[g]) limit exemption to the subcriticality requirements (10 CFR 71.55[b]) to contents enriched to a maximum of 5 percent. For feed material enriched to greater than 5 percent, the package would have to be evaluated with water inside the cask, or the applicant would need to request an exemption (U.S. NRC, 2020f). Furthermore, U.S. Department of Transportation (DOT) regulations (49 CFR 173.420) require that UF_6 packages be designed according to American National Standards Institute Standard N14.1, which only applies to enrichments up to 5 percent. Use of the 30B cask to transport enriched UF_6 in the United States will require advanced planning to assure transport capability when needed.

As mentioned above, feedstock material in either oxide or metallic form may need to be shipped from the enrichment facility to the fuel fabrication facility. The three currently available packages for these shipments, the ES-3100, TN-BGC1, and Versa-Pac VP-55, have limited capacities. It is anticipated that packages in the hundreds of kilograms will be needed to meet the demand for HALEU (Redmond and Ashkeboussi, 2022). Once fresh fuel with initial enrichments greater than 5 percent has been fabricated and needs to be shipped to the reactor in a fresh fuel packaging, criticality assessments will need to be made for the transport package. The U.S. NRC has stated that there is a lack of critical experiments[12] providing data to support criticality benchmarking analyses (U.S. NRC, 2020f) for transport packages containing material enriched above 5 percent. The U.S. NRC recommends addressing this issue by

- performing new critical experiments,
- extrapolating existing critical experiment data using uncertainty or sensitivity analyses,
- increasing safety margins due to lack of validation analyses, or
- a combination of the above three methods.

In support of the DOE advanced reactor initiatives, assessment of transport challenges was recognized at the INL Workshop on HALEU discussed in Chapter 4. As part of this workshop, industry feedback indicated the need for DOE support in development and licensing of a HALEU transport container (Caponiti, 2020a).

For storage of spent fuel containing HALEU, technical issues associated with criticality and higher temperatures will need to be evaluated. Criticality design generally relies on neutron poisons in either soluble or solid forms (or both) in the spent fuel pool or fixed poisons in the basket structure of dry storage containers (U.S. NRC, 2020e). Pool and dry storage of spent fuel with enrichments of greater than 5 percent will need to be assessed to quantify the impact on current designs and operations as they relate to maintaining subcriticality during storage. Thermally, the higher burnups expected for these fuels will result in higher temperatures in the spent fuel. Management of these heat loads may require longer in-pool storage before transferring to dry storage. In dry storage and drying operations during packaging, assessments will be needed to ensure peak temperatures of the fuel remain below designated limits.

For the back-end transport, all forms of spent fuel containing HALEU will require assessments prior to transport. All transport systems must be evaluated under hypothetical accident conditions to ensure that the used fuel payload will remain subcritical. With initial enrichments of up to 19.75 percent, transport packages may need to be redesigned to assure that subcriticality thresholds are maintained. HALEU spent fuel will also have higher burnups, which will result in higher heat and radiation loads. Thermal and radiation shielding analyses will be required and also may result in package redesign. In particular, for thermal reactors,[13] neutron dose rates may increase significantly due to curium-242 or -244 content (IAEA, 2020d).

[12] Critical experiments use combinations of different geometries, fissile material (fissile uranium or plutonium), and moderator (water, plastic, etc.) to achieve a k_{eff} (effective neutron multiplication factor) of exactly 1.0. These data are then used to evaluate the accuracy of the code used to evaluate criticality safety by modeling a large number of critical experiments to determine statistically whether there is a bias for the code (i.e., whether the code consistently calculates a k_{eff} higher or lower than 1.0) (U.S. NRC, 2020f).

[13] The sentence was modified following a prepublication version of the report to indicate the applicability to thermal reactors only, and not fast reactors.

5.6.2.2 Uranium Oxide Fuels for Integral Pressurized Water Reactors (iPWRs)

The NuScale iPWR Small Modular Reactor is identified as an advanced reactor concept and has progressed in U.S. NRC licensing reviews by receiving design certification in August 2020 (Reyes, 2021). The fuel technology for this small modular reactor is a standard LWR 17×17 PWR fuel design using UO_2 fuel pellets (<5 percent–enriched) housed in M5 cladding. As such, storage and transportation licensing is expected to be consistent with existing LWR applications and does not represent a change in licensing risk. Licensing experience for existing LWR storage and transportation systems suggests that review of new applications for similar small modular reactor systems should be consistent with this past experience.

Similarly, secondary waste facilities for low-level wastes, including GTCC, are envisioned at the plant site. Dry storage systems for these waste streams are envisioned to be consistent with existing licensed GTCC and low-level waste designs. As mentioned earlier, waste volumes may be a concern as multiple units per reactor island are envisioned. A careful assessment of these impacts is required to develop a clear picture of the scope of the waste stream.

5.6.2.3 TRistructural ISOtropic (TRISO) Particle Fuel and Graphite Materials

All of the advanced reactor systems proposing to use TRISO envision using HALEU source material with up to 19.75 percent–enriched uranium. From a storage perspective, the spent TRISO fuel provides a robust containment where the graphite cover acts as a protective barrier similar to the metal cladding used in LWR systems. Existing storage systems have licensed TRISO spent fuel in prismatic graphite blocks at the Fort St. Vrain Independent Spent Fuel Storage Installation (ISFSI) (Hall et al., 2019b). For TRISO in pebble form, storage can be designed using dry canisters with overpacks, similar to conventional dry storage systems for LWR fuel. However, proposed advanced designs will require additional technical assessments (Hall et al., 2019b). First, maintenance of subcriticality margins of TRISO fuel with higher enrichments (>5 percent) and higher burnups will need to be verified. Second, higher decay heat may require additional thermal analyses to verify thermal performance with the higher burnup spent fuel. Finally, fluorine salt–cooled reactors may leave residual salt on the TRISO particles, which may result in radiolysis generating fluorine gas, creating issues concerning both safety and corrosion on the containment package.

In a study of DOE-owned nuclear fuel, the Nuclear Waste Technical Review Board (NWTRB) identified a number of technical issues related to the storage and disposal of the TRISO fuel generated by the Fort St. Vrain reactor (NWTRB, 2017): The extended storage of carbide-based fuels may require careful attention to aging-management issues in order to successfully recover and treat these fuels. Degradation and corrosion processes associated with a carbide-based fuel are different enough from those of an oxide-fuel that they require special investigation. In particular, careful attention must be paid to the possibilities of (1) gas generation; (2) mechanisms of waste form degradation, dissolution, and precipitation of specific radionuclides; (3) solubility limits of radionuclides; and (4) sorption/desorption reactions of radionuclides onto degradation products of the waste package components, to name a few. These identified technical issues illustrate the importance of considerations across operational boundaries—in this case, spanning storage and disposal operations.

As discussed above, management of residual and used graphite will need to be addressed in terms of degradation characteristics in long-term storage environments. Reactors using graphite as moderators and reflectors will produce large quantities of spent graphite over the course of reactor life; therefore, the development of graphite disposal technologies needs attention.

From a transportation perspective, spent TRISO fuel does provide a robust primary containment that will likely survive normal and loadings for hypothetical accident conditions as defined in 10 CFR 71. The Fort St. Vrain transportation cask (TN-FSV) has been licensed to transport TRISO fuel (Hall et al., 2020). Structural, thermal, criticality, and shielding considerations satisfy current 10 CFR 71 regulations. However, the proposed higher initial fuel enrichments suggest that criticality analyses will need to be conducted to assure subcriticality margins in the 10 CFR 71.55(e) nonmechanistic loading requirement to ensure payload subcriticality. Alternatively, a moderator

exclusion approach could be assessed that would include a double seal to prevent water ingress. Higher burnups will likely require packaging thermal analyses to ensure threshold temperatures are not exceeded on the cask surface.

In general, design of storage and transportation systems for TRISO spent fuel and associated secondary waste streams should be achievable under the current regulatory structure. However, it will be important to identify and assess higher enrichments and burnups, as well as degradation characteristics of these wastes that may affect the functional characteristics of the designed storage and transportation systems.

5.6.2.4 Metallic Fuels and Materials

TerraPower, ARC, LeadCold, and Oklo have proposed advanced fast reactor designs that use metallic fuel with sodium or lead as the liquid coolant (DeWitte, 2021; Hejzlar, 2021; Neider, 2021; Sackett and Arthur, 2021). Metallic fuel proposed for advanced reactors requires special consideration. The storage experience at INL indicates that issues that may affect the integrity of sodium-bonded fuel have not been fully addressed (Hall et al., 2019b). Initial storage (pool or dry), followed by long-term storage in a confinement package, needs to demonstrate a long-term ability to resist moisture degradation of both the confinement package barrier and the fuel components. Long-term performance characteristics of any proposed sodium-bonded fuel will need to be developed, taking into account the storage environment (temperature, humidity, etc.) along with chemical, radiological, and mechanical interactions between the various materials (metallic fuel, sodium layer, and stainless-steel cladding) that make up the fuel and surrounding confinement boundary. Individual fuel types with specific storage system designs will need confirmation testing and analyses to validate degradation characteristics of the fuel and storage system in order to assess performance against the regulations. While still in the R&D phase, Oklo is partnering with INL through an Advanced Research Projects Agency-Energy project to develop a bondless metallic fuel (DeWitte, 2021). All sodium-bonded spent fuel will have to undergo some form of cleaning/stabilization/reprocessing in order to make it safe for long-term storage. Storage at INL includes fuel that was chemically treated to deactivate the sodium and then was converted to ceramic or metallic HLW. The INL inventory also includes untreated sodium-bonded metallic spent fuel. This indicates the difficulty in managing the spent fuel as a waste form once the fuel is removed from the reactor. The operational difficulties arising from stabilization of this form of spent fuel to prepare it for long-term storage, transportation, and eventual disposal point to the need for reliable cost estimating to treat the spent fuel. Treatment estimates then need to be factored into the total life-cycle costs in order to provide a realistic perspective of total costs associated with the back-end management of the spent fuel.

These advanced fuels also propose using HALEU to increase initial enrichments up to 19.75 percent. As with TRISO fuel, criticality assessments will need to be conducted. For designs with high burnup, thermal analyses will have to be performed to verify thermal performance.

Secondary wastes from the metal coolant need to be considered. Current planning and design is such that it is too early to provide specific assessments concerning these wastes. However, development of plans to safely deal with the storage of these wastes need to be addressed as early as possible. The manner in which residual and excess metal is processed, cleaned, and stored needs to be developed and defined for these systems. Waste forms and packaging systems then will need to be assessed in terms of degradation characteristics relative to the storage environments.

There is a limited history of transporting sodium-bonded metallic fuel in the DOE complex (Hall et al., 2020). The T-3 cask was used to ship FFTF fuel from Hanford, Washington, to Idaho, and EBR-II fuel was shipped intrasite at INL facilities using both the NAC LWT and TN-FSV casks. These previous shipments provide evidence of licensability for transport cask designed to ship metallic spent fuel. As with storage considerations, the condition of the fuel and any confinement barrier will need to be confirmed prior to shipment in order to assess its ability to withstand both normal and hypothetical conditions of transport. As with other HALEU fuel, higher initial enrichments will require assessment to assure criticality margins meet the regulations under criteria for hypothetical accident conditions. Design burnups as high as 150 MWd/MTU (megawatt-day/metric ton of uranium) are proposed, and thermal analyses will need to be conducted to verify thermal performance of the fuel and its packaging against transport conditions.

Degradation characteristics for transportation are less of an issue than for storage, in that transportation is a short-term event. However, the integrity and condition of the waste form must be verified prior to shipping in order to assure the integrity of the waste during the shipping event.

5.6.2.5 Molten Salt Liquid Fuels

Terrestrial Energy, ThorCon, Flibe, TerraPower, and Moltex all have advanced reactor designs that use fuel as part of a molten salt mixture (Jorgensen, 2021; Latkowski, 2021; LeBlanc, 2021; O'Sullivan, 2021; Sorensen, 2021). Terrestrial Energy, ThorCon, and Flibe use a fluoride salt, and TerraPower and Moltex use a chloride salt. In general, there is a limited experience base for designing, licensing, constructing, and operating the back end of MSR fuel cycles.

Planning for back-end management of spent fuel and secondary wastes has been initiated with some fairly detailed specifics on waste form characteristics and on the secondary waste stream definition. All concepts involve some form of recycling, conversion, and/or stabilization to process the wastes into forms that can maintain integrity during long-term storage, transportation, and disposal conditions.

From a storage and transportation perspective, licensing of systems that provide confinement/containment, shielding, criticality control, and thermal management will have to be assessed via the current regulatory structure. Given the state of the development of these types of advanced reactors and the lack of operational experience of commercial-scale reprocessing, R&D is needed beyond conceptual thinking to better define waste form isotopics and concentrations, hazardous constituent assessments, degradation characteristics, and overall performance relative to the storage and transportation environments to which they will be subjected. Numerous options are available for processing, from converting the spent fuel to a stable solidified block, to separating the fuel from the salt and reprocessing and cleaning of the salt for reuse. However, little practical experience exists to identify potential issues with implementation at scale. The thermal spectrum reactors also use graphite for a moderator, which will have to be managed as a secondary waste. Issues such as the presence of carbon-14 and daughter products of xenon and krypton will have to be addressed. Much work is needed to develop these concepts to a point where an assessment of the impact on storage and transportation can be properly assessed.

5.6.3 Summary: Storage and Transportation of Advanced Fuels

The current storage and transportation regulatory structure will accommodate review and licensing of advanced reactor fuel cycle materials and systems. As has been shown in the past, considerably more work may be needed in some areas to demonstrate compliance with the regulatory criteria. Because many of the proposed fuels will use HALEU with initial enrichments up to 19.75 percent, criticality becomes an especially important safety function that may require additional testing and analyses to demonstrate compliance with the regulations for these advanced material waste forms. Validation that criticality margins will be satisfied may need additional testing to develop a database of the physical, chemical, and isotopic makeup of the waste form, along with analyses to validate the criticality margins.

Thermal management of proposed high burnup fuels may require quantification of the thermal environment and response characteristics of waste materials and packages to this environment.

For metallic fuels, R&D is ongoing to define how best to prepare the spent fuel for long-term storage in a way that best stabilizes the spent fuel. The wet and dry storage experiences at INL (Hall et al., 2019b) indicate that issues associated with degradation of the fuel need to be addressed. Dissimilar metals, sodium, moisture, and temperature all act as catalysts in potential degradation processes. For transportation of these fuels, degradation is less of an issue because of the short duration of the transportation event. However, this does point to the need to know the condition of the spent fuel prior to shipment. Typical plans call for the canisterization of the spent fuel for storage. This package would then be used for transport after storage, making verification of fuel integrity difficult. The spent fuel could be in storage for many decades or conceivably a century; thus, it is essential to characterize the spent fuel's condition over this long time.

Management of liquid spent fuel is still in a development stage, with significant issues needing to be

addressed on how best to treat the spent fuel. As this technology becomes better defined, waste forms will need complete characterization of physical form, chemical composition, and radiological characteristics. Based on these characteristics, waste form and packaging degradation processes will need to be addressed to understand how the waste form may degrade in long-term storage conditions. The waste form characteristics, along with the understanding of degradation processes, will provide the information needed for licensing storage and transportation systems.

Secondary wastes from these advanced fuel concepts that are not generated in LWR operations include graphite, salt, lead, and sodium. This inventory includes hazardous wastes, along with GTCC and low-level wastes. While hazardous wastes are not specifically covered in 10 CFR 71 and 72, DOT regulations (49 CFR 173) will need to be followed. Packaging of these materials for storage and transportation is not seen as a difficult licensing problem, but work is needed to define the waste form characteristics, along with quantities generated.

Storage and transportation operations in the context of this study represent a link between reactor power generation and disposal of resultant wastes. While important, these operations are not the primary drivers in the licensing, technical, or economic decision-making required for the development of advanced fuel cycles. No "show-stopper" storage and transportation issues have been identified that would indicate a reason to second guess the development of any of these proposed concepts. However, significant work is needed in some areas to properly characterize waste forms in a way that quantifies response characteristics to the storage and transportation environments. *In some cases, this will require significant testing and R&D analyses to properly understand waste form and packaging performance in relation to the storage and transportation environment to which they are subjected. Management and storage of these new spent fuels will in some cases entail additional costs significantly above those associated with existing LWRs.*

In general, the NWTRB determined regarding DOE-owned spent nuclear fuel, "*Given the uncertainty in how long DOE SNF [spent nuclear fuel] will be stored prior to disposal, the Board finds that having the ability to measure and monitor conditions of the SNF inside canisters, the external surfaces of canisters, and the storage facility itself during future storage is an important consideration in designing, developing, and deploying new DOE storage systems, such as a DOE standardized canister, and for new packaging and storage facilities*" [emphasis added] (NWTRB, 2017).

5.7 AN OVERVIEW OF DECONTAMINATION AND DECOMMISSIONING OF NUCLEAR POWER PLANTS

5.7.1 Background Status

Decontamination and decommissioning (D&D) of nuclear facilities around the world has a long history to support an ever-increasing demand, as the global nuclear industry is facing a fleet of aging plants and early plant shutdowns for economic reasons. The 414 reactors operating worldwide as of March 2022 have an average age of 31 years (WNISR, 2022).

The average age of the 93 currently operating reactors in the United States is 39 years (EIA, 2021c), and 21 commercial power plants have been decommissioned or are currently decommissioning (U.S. NRC, 2021d). The United States' substantial background in D&D began with the Shippingport plant in Pennsylvania. As the world's first full-scale commercial nuclear power plant, Shippingport began producing power in 1957 and continued producing power until its shutdown in 1982 (Crimi, 1995). The site was released for unrestricted use after D&D activities were completed in 1989. However, the Beaver Valley, Units 2 and 3, nuclear generating station located on the same site is still operating. The Shippingport reactor vessel was shipped by barge to the Hanford Site for burial. The spent nuclear fuel is stored at the Hanford Site Canister Storage Building in 18 canisters (NWTRB, 2020). Low-level wastes were treated on site, packaged, and shipped to low-level waste burial sites.

A more recent example of successful D&D is the Trojan Nuclear Power Plant. The Trojan plant, located on the Columbia River in Oregon, operated from 1975 to 1992, and D&D was completed in 2008 (Nuclear Decommissioning Collaborative, 2022). The reactor vessel was barged up the Columbia River for burial at the Hanford Site. Different from the Shippingport solution, spent nuclear fuel at Trojan is stored on site at a licensed Independent Spent Fuel

Storage Installation (ISFSI) in 34 dry storage casks (Oregon DOE, n.d.). Without a national solution to a repository at this time, this has become a common way for utilities to deal with their spent nuclear fuel at shutdown sites.

As discussed in Section 5.1, low-level waste generated from current LWR power reactor operations is treated, stored, and shipped to one of four licensed low-level waste disposal facilities in the United States.

The operational legacy of D&D provides a basis for consideration of how best to manage both high- and low-level wastes from advanced reactors and fuel cycles. This experience, with a focus on LWR operations, also provides a basis for identifying gaps that will have to be considered in management of D&D associated with advanced reactors and fuel cycles.

5.7.2 International Guidance and U.S. Regulatory Status

The IAEA, which is not a regulator, provides international guidance on D&D; it formally began work in this area in 1973, publishing its first document on D&D issues in 1975 (IAEA, 1999b). Since then, dozens of IAEA reports have been published assessing the technological feasibility of safely conducting D&D operations. These assessments include evaluating specific waste streams, treatment of wastes, and methods to conduct the actual dismantling operations and sizing of the debris for further processing, handling, and packaging. Two reports in the form of "TECDOCs" were published covering information exchange, experiences in actual D&D activities, and lessons learned from these activities (IAEA, 1989, 2018a). More recently, the IAEA published a state-of-the-art report for specific technology processes for decontamination and dismantling of radioactive facilities (IAEA, 1999b). This work reflects an ongoing effort to assess D&D of nuclear facilities in light of the continual D&D activities in the industry as plants continue to age and go into shut-down status.

The IAEA has also published a safety guideline (IAEA, 2018a) that provides guidance for meeting safety requirements applicable to decommissioning nuclear facilities. This guide is more of an administrative roadmap for regulators and licensees to follow in decommissioning planning as opposed to the technical documents identified previously. The safety guide also provides a comprehensive list of other IAEA publications relevant to decommissioning.

In the United States, the U.S. NRC continues to assess D&D activities of licensed plants that have shut down. Currently, three decommissioning strategies are available to licensees: DECON, SAFSTOR, and ENTOMB. DECON refers to dismantling and removal of all plant structures and radioactive contaminated material to a level that would permit release of the property. SAFSTOR allows for maintenance of the facility prior to decommissioning while radioactivity of spent fuel and contaminated materials decays and decommissioning funds are amassed. Once a plant is ready, it moves from SAFSTOR to DECON. The plant is then dismantled and decontaminated. ENTOMB allows for on-site burial of encased radioactive materials and components in structurally sound containment barriers, such as concrete. The property can be released for restricted use after radioactive decay reaches a certain level defined in the license. To date, no U.S. NRC–licensed nuclear power plants have used this option (U.S. NRC, 2020m). In the SAFSTOR and in the DECON strategy as currently practiced, spent nuclear fuel is stored on-site at a licensed ISFSI facility. U.S. NRC regulations associated with D&D are promulgated in 10 CFR 20(E) and 10 CFR 50.75/82/53/96.

As of December 2021, 26 power reactors were under DECON or SAFSTOR licensing protocols (U.S. NRC, 2022c). Thirteen reactors are using the DECON strategy: Humboldt Bay 3, San Onofre 1/2/3, Zion 1/2, LaCrosse, Crystal River, Fort Calhoun, Oyster Creek, Pilgrim, Vermont Yankee, and the N.S. Savannah. Thirteen reactors are using the SAFSTOR strategy: GE EVERSR, GE VBWR, Millstone, Dresden, Duane Arnold, Fermi 2, Indian Point 1/2/3, Peach Bottom, Three Mile Island 1/2, and Kewaunee. Furthermore, 10 power reactors have completed decommissioning: Rancho Seco, Fort St. Vrain, Haddam Neck, Maine Yankee, Yankee Rowe, Big Rock Point, Shoreham, Trojan, Saxton, and Pathfinder.

This level of D&D activity at U.S. NRC–licensed facilities provides reasonable assurance that D&D can be carried out properly at nuclear facilities, with the site returning to unrestricted use pending completion of the D&D. For the U.S. fleet of LWRs using low-enriched uranium oxide fuel, plants can be properly dismantled, separating the uncontaminated components and materials from the contaminated materials and spent nuclear fuel. Uncontaminated materials can be recycled or disposed of in landfills. *For all U.S. sites that have fully decommissioned, there is still*

no pathway to disposal for their spent fuel, and it remains on site. As a result, the largest challenge to the ability to return sites completely to unrestricted use is to establish one or more geologic repositories for spent fuel disposal.

5.7.3 Advanced Reactors and Fuel Cycles

The proposed technologies for advanced reactors include fuel and materials that are markedly different from the current LWR fleet in the United States. Fast reactors using sodium, lead, or molten salt coolants coupled with metallic, TRISO, and liquid fuels in molten salt will create challenges in terms of D&D technologies and operations. As opposed to LWR operations, this new generation of reactors will create secondary wastes that may be particularly challenging in terms of stabilizing, packaging, storing, and disposal. Sodium and lead coolants, sodium-bonded fuels, molten salts, and graphite moderators and reflectors will create wastes that will need special attention and probably additional R&D to safely process and stabilize, adding costs to storage not experienced with LWRs. Characterization, processing, and volumes of materials after reactor shutdown remains unclear at this point. These waste streams point to the gap in the operational legacy mentioned earlier.

The IAEA has published a preliminary assessment of expected types of wastes generated from advanced reactors (IAEA, 2019a). This document provides a good general assessment of the specific types and volumes of wastes that can be expected from advanced water-cooled reactors; gas-cooled reactors; liquid metal reactors; and dedicated actinide burners, such as molten salt reactors.

Noncontaminated components of advanced reactors can be disposed of via existing approved methods. As with LWR spent fuel, the spent nuclear fuel from advanced reactors will have to be stabilized, packaged, and stored until a disposal path is available for the United States. Secondary wastes generated in these proposed advanced reactors will need careful consideration as to how they can best be managed as they are generated and removed from the reactor system. In some cases, R&D will need to be performed to understand degradation processes of these wastes and how they will interact with their environments (e.g., long-term storage in licensed dry storage systems and assessment of activation products in graphite). Wastes generated from spent fuel processing will also have to be identified and addressed (e.g., cleaning of TRISO fuel used in molten salt coolants).

Lessons learned from existing experiences can provide valuable insights into specific issues associated with managing some of these materials (Hall et al., 2019b). Notably, for many of the past "advanced" designs in the United States, the spent fuel remains incompletely treated. This is true of the sodium-bonded spent fuel from various sodium-cooled fast reactor experiments, as well as HTGR spent fuel (from Fort St. Vrain) and spent fuel from the Molten Salt Reactor Experiment at Oak Ridge.

For example, INL has ongoing experience in the long-term storage of sodium-bonded metallic fuel. Issues with early pool storage have led to corrosion of some of the fuel. While this fuel is being treated and moved to dry storage, insights into early management of this fuel once it is removed from the reactor can provide important guidance for proper processing and storage to prevent such problems in advanced reactor operations.

The Fermi-1 sodium-cooled breeder reactor has been shut down and is in the SAFSTOR licensing process under the U.S. NRC, with final site closure slated for 2032. Currently, all spent nuclear fuel, bulk sodium, and the reactor vessel, as well as other major components have been removed from the Fermi site (U.S. NRC, 2021e). The spent fuel and blanket subassemblies were shipped to INL for long-term storage. The secondary nonradioactive sodium was shipped to a commercial chemical company for reuse (U.S. NRC, 2021e). The primary sodium was stored on-site in tanks and 55-gallon drums until it was shipped to INL (Argonne-West) in the early 1970s. The sodium was treated for long-term storage at INL (Sherman and Knight, 2005). INL is still in the process of cleaning and remediating these storage tanks (U.S. NRC, 2020i). This example points to the historical difficulties in dealing effectively with sodium-bearing wastes, which will need to be addressed through additional R&D.

Fort St. Vrain, an HTGR, has been fully decommissioned except for a licensed ISFSI on-site to store its spent fuel. This provides one example of a licensed power-producing advanced reactor that has been decommissioned according to U.S. NRC regulations. Many of the plant's components and systems have remained in place as the facility was converted to a natural gas plant. With the exception of the spent fuel, components categorized as low-level and secondary low-level wastes were treated and appropriately disposed in low-level waste disposal

facilities (Fisher et al., 1996). As with LWR plants, the challenge for the United States is to operationalize a final pathway for the spent fuel for HTGRs.

As a secondary waste, the use of graphite for moderation and as a reflector is a component of several of the proposed advanced reactors; this will result in a sizable amount of waste that will need to be disposed of after reactor operations cease. The IAEA has recognized the significance of this problem and in 2006 published a TECDOC identifying the technical issues and challenges in managing graphite as a back-end product of reactor operations (IAEA, 2006). At the time of the publication, there were more than 230,000 MT of graphite needing disposition. A single full-size reactor can result in up to 3,000 MT of graphite. Tritium, carbon-14, and chlorine-36 are all generated and need to be carefully considered in any back-end management plan. The IAEA also recently designated Collaborating Centre in France for graphite reactor decommissioning (Kilochytska, 2021), which will fund R&D to advance technologies in the management of graphite disposition.

Other secondary wastes—such as molten salt used for cooling, processing of liquid fuels, and off-gassing—will need to be properly characterized and addressed in terms of back-end management in order to provide a safe, complete fuel cycle all the way to disposal. From a SAFSTOR perspective, advanced fuels using HALEU enriched up to 19.75 percent will need to be stored pending a resolution on a U.S. repository. Issues associated with criticality and thermal management due to the high burnup of most of the fuel designs will need to be considered early in the design process.

For advanced reactor concepts that are considering reprocessing to close the fuel cycle, the West Valley Demonstration Project (WVDP), classified as a Complex Decommissioning Site by the U.S. NRC, provides an example of D&D challenges for a spent fuel reprocessing facility (U.S. NRC, 2021f). The West Valley Plant operated from 1966 to 1972. It is still in an active state of D&D, with no firm date for completion (estimated completion date, 2040). Under the 1980 WVDP Act, Congress directed DOE to assume possession of the major components of the facility, including the reprocessing plant, the U.S. NRC–licensed disposal area, the high-level waste tanks and waste lagoons, and the aboveground storage areas. Furthermore, the Act authorized DOE to "solidify, transport and dispose of HLW [high-level waste] that exists at the site, dispose of LLW [low-level waste] and transuranic waste produced by the WVDP, and decontaminate and decommission facilities used for the WVDP in accordance with requirements prescribed by [the U.S.] NRC" (WVDP Act, 1980 [Public Law 96-368]). The liquid high-level waste produced from reprocessing has been vitrified, packaged, and stored on-site. While the vitrification facility has been demolished, the Main Plant Process Building is conducting deactivation activities to complete removal of fixed plutonium in order to move forward with open air demolition. WVDP costs up to 2013 were $2.25 billion. Estimated costs from 2013 to completion were an additional $2.73 billion (Rieman, 2013).

For active designers of advanced reactors and fuel cycles, it is important to address these issues up front in order to provide a complete picture of what will be required from a technical, safety, and economic standpoint in fielding these systems and in providing responsible back-end solutions to waste disposition.

6

Nonproliferation Implications and Security Risks

This chapter reviews the nonproliferation implications and security risks of the fuels and fuel cycles associated with advanced nuclear reactors. It responds to the charges in the statement of task that call for assessments of high-assay low-enriched uranium (HALEU), uranium-plutonium mixed oxide fuel, and advanced fuel cycles that require separating plutonium from spent fuel; examination of nuclear material accounting and control, as well as containment, surveillance, monitoring, and timeliness of detection of diversion; and suggestions for how these nonproliferation implications and security risks can be addressed by International Atomic Energy Agency safeguard activities.

In this chapter, the committee provides the summary, findings, and recommendations up front (Section 6.1); describes relevant aspects of nonproliferation, safeguards, and nuclear security, including transportation security (Section 6.2); and examines the nonproliferation implications and security risks of fuels and fuel cycles for various advanced reactors, with particular attention to HALEU because of the interest expressed by most advanced reactor developers in using this material in their fuels (Section 6.3).

6.1 CHAPTER 6 SUMMARY, FINDINGS, AND RECOMMENDATIONS

Deployment of advanced reactors and their supporting fuel cycles will involve the production, transportation, storage, and irradiation of nuclear materials with significantly different characteristics from those of the current U.S. light water reactor (LWR) fleet, which is fueled with low-enriched uranium. These differences could impact nuclear proliferation[1] and nuclear terrorism[2] concerns posed by advanced reactors. Evaluating advanced reactor fuel cycles with respect to their potential nuclear proliferation implications and nuclear terrorism risks (referred to as "safeguards" and "security risks") is warranted. It is also important to assess the feasibility and efficacy of ways to mitigate these risks by improving and strengthening the technical and institutional measures needed for nuclear material safeguards and physical protection.

Addressing these issues first requires identifying the features of advanced reactor fuel cycles relevant to safeguards and security. These include the following considerations: Advanced reactors may variously use special nuclear materials, such as HALEU, plutonium, minor actinides, or thorium and uranium-233, either separately or in mixtures. As discussed in Chapter 4, most advanced reactor designs propose using HALEU fuel, which, given

[1] *Nuclear proliferation* is defined as diversion or misuse of civil nuclear materials by a state to produce nuclear weapons.

[2] *Nuclear terrorism* is defined as acquisition of nuclear materials by substate actors seeking to build improvised nuclear explosive devices.

that its material attractiveness is higher than low-enriched fuel, would require additional security considerations throughout the fuel cycle and potentially a new safeguards framework to mitigate proliferation concerns. Fuels for advanced reactors may also have nuclear material concentrations and chemical and physical forms that differ from LWR fuel. Advanced reactors may use different refueling cycles than LWRs, ranging from online refueling to long-lived once-through cores, both of which present unique safeguards and security challenges. Deployment of small modular reactors or microreactors would increase the number of units and sites requiring safeguards and security measures for a given amount of power capacity compared with large reactors—which could increase the burden on international safeguards inspectors and domestic security resources. One of the most important factors is whether fuel cycles are open (using interim storage and eventual direct disposal of spent fuel) or closed (involving reprocessing and reuse of separated nuclear materials).

With respect to the main focus of this report, the deployment of advanced reactors and fuel cycles in the United States, direct proliferation by the United States is not a concern, as it is a nuclear weapon state. Nonetheless, safeguards obligations on domestic facilities that the United States could assume under its voluntary offer agreement with the International Atomic Energy Agency (IAEA) would have indirect benefits by providing test beds for IAEA safeguards approaches for new technologies that may be deployed in other countries and would set an example for international cooperation.

Nuclear terrorism, however, is a concern within the United States and therefore must be addressed in the physical protection and emergency response plans for domestic nuclear facilities. As the 9/11 attacks demonstrated, various elements of the U.S. critical infrastructure could be a target for attack by terrorist organizations and require a high level of physical protection. Today, such groups include a range of substate actors, both foreign and domestic in origin, as well as those receiving state support. These threats can be carried out by insiders and/or external adversary groups, and could involve both physical attacks and cyberattacks. Current security requirements with regard to both theft and radiological sabotage may need to be modified to address the specific characteristics of various advanced reactors and fuel cycles.[3]

The committee makes the following findings and recommendations regarding the nonproliferation implications and security risks of advanced reactors and fuel cycles:

Finding 19: Expanding the global use of high-assay low-enriched uranium (HALEU) would potentially exacerbate proliferation and security risks because of the potentially greater attractiveness of this material for nuclear weapons compared with the low-enriched uranium used in light water reactors. The increased number of sites using and states producing this material could provide more opportunity for diversion by state or nonstate actors.

Recommendation M: The U.S. National Nuclear Security Administration, in coordination with the U.S. Department of Energy's Office of Nuclear Energy, should assess proliferation and security risks associated with high-assay low-enriched uranium (HALEU) and its potential for expanded global use. In parallel, the U.S. government should foster an international effort, which could be facilitated by the International Atomic Energy Agency, to examine and address these risks.

Finding 20: All of the advanced reactor fuel cycles will require rigorous measures for safeguards and security commensurate with the potential risks they pose. Issues requiring special attention include the following:

- Material accountancy (i.e., tracking and quantification) is more difficult for molten salt and pebble-bed technologies than for reactor systems that use stationary solid fuels because of the technical challenges in performing measurements with online fuel and bulk-handling facilities. Containment and surveillance will also be more challenging to implement for these types of reactors. Thorium/uranium-233 fuel cycles require development of safeguards technology because of the large number of variants in their systems. Moreover, safeguards tailored to traditional uranium/plutonium fuel cycles are not applicable to these systems.

[3] A detailed evaluation of radiological sabotage is beyond the scope of this report.

- Fuel cycles involving reprocessing and separation of fissile material that could be weapons usable pose greater proliferation and terrorism risks than the once-through uranium fuel cycle with direct disposal of spent fuel, as the separated fissile material would not be uniformly mixed with highly radioactive fission products. Separated, potentially weapons-usable materials could include fissionable materials other than the "traditional" special nuclear materials of highly enriched uranium, plutonium, and uranium-233. Thus, for these closed fuel cycles, specific safeguard technologies will likely be required to meet the International Atomic Energy Agency's goal of timely detection.

Recommendation N: The U.S. government should support the International Atomic Energy Agency's (IAEA's) development and application of effective safeguards for advanced reactor technologies by authorizing, via the U.S. interagency process, IAEA access through the eligible facilities list, especially to those advanced reactor systems for which the IAEA does not currently have safeguards experience. Developers of these types of advanced reactors and fuel cycle facilities should provide facility information to the IAEA to help with integration of safeguards considerations into the design process.

Recommendation O: The U.S. Nuclear Regulatory Commission should initiate a rulemaking to address the security and material accounting measures for high-assay low-enriched uranium (HALEU) and other attractive nuclear materials that may be present in advanced reactor fuel cycles.

6.2 BACKGROUND ON NONPROLIFERATION, NUCLEAR MATERIAL SAFEGUARDS, AND NUCLEAR SECURITY

6.2.1 Nuclear Nonproliferation

The Treaty on the Non-Proliferation of Nuclear Weapons (NPT), which was signed in 1968, entered into force in 1970, and was extended indefinitely in 1995, seeks to globally "prevent the spread of nuclear weapons, to promote cooperation in peaceful uses of nuclear energy and to further the goal of nuclear disarmament"[4] (DOS, 2010, n.d.; IAEA, n.d.-a; NPT, 1968; UN, n.d.-a). The three goals of the NPT—nonproliferation of nuclear weapons, nuclear disarmament, and peaceful use of nuclear energy—are described in detail in such sources as IAEA (1970, n.d.-a) and UN (n.d.-a).

6.2.2 Safeguards

The goal of safeguards is verifying that states are in compliance with treaty obligations prohibiting the diversion of civil nuclear materials and technologies for nuclear weapons. After states sign Comprehensive Safeguard Agreements[5] (CSAs) or Voluntary Offer Agreements[6] (VOAs), the objective of the IAEA is to verify a state's safeguards compliance in order to ensure that "the nuclear material, non-nuclear material, services, equipment, facilities and information specified and placed under safeguards are not used for the manufacture of nuclear weapons or any other nuclear explosive devices or to further any military purpose" (IAEA, 2002). In this regard, the technical objective is specified as "the timely detection of diversion of significant quantities of nuclear material from peaceful nuclear activities to the manufacture of nuclear weapons or of other nuclear explosive devices or for purposes unknown, and deterrence of such diversion by the risk of early detection" (IAEA, 2002).

[4] There are 191 states partied to the NPT, with only 5 recognized as nuclear weapon states: the United States, the United Kingdom, China, France, and Russia. Only these states are legally allowed to possess nuclear weapons under the NPT, and they have agreed to make good-faith efforts to disarm in the future. The NPT is open to any UN member state, provided they sign as a non–nuclear weapon state. Israel, India, Pakistan, and South Sudan are the only 4 of the 193 UN member states that are not party to the treaty. Palestine and the Holy See are observer states and thus technically party to the NPT, although they are not UN member states. North Korea claims to have left the treaty in 2003, but this action has not been officially recognized by all states party to the NPT.

[5] Each non–nuclear weapon state party to the NPT is required to sign a CSA with the IAEA.

[6] Nuclear weapon states can voluntarily offer to place selected nuclear facilities under IAEA safeguards via VOAs.

Safeguards procedures focus on ensuring peaceful uses of nuclear materials and technologies and detecting the diversion of significant quantities of nuclear materials in a timely manner. General tools used for safeguards include material accountancy (using various nuclear materials measurement techniques and equipment); containment and surveillance measures; process measurements, inspections, and tamper-indicating marking; monitoring (unattended and/or remote); and environmental sampling.

IAEA inspectors conduct various visits and inspections at facilities or other locations to verify a state's compliance with the CSA or VOA obligations or to determine a state's noncompliance. Noncompliance regarding the CSA/VOA obligations might involve diversion of nuclear material and undeclared nuclear material or activities. Examples of diversion of nuclear material include "(1) the undeclared removal of declared nuclear material from a safeguarded facility; (2) the use of a safeguarded facility for the introduction, production, or processing of undeclared nuclear material, e.g. the undeclared production of highly enriched uranium in an enrichment plant; and (3) the undeclared production of plutonium in a reactor through irradiation and subsequent removal of undeclared uranium targets" (IAEA, 2002). The IAEA has produced additional safeguards reports and recommendations specific to particular types of nuclear facilities or particular parts of the nuclear fuel cycle; see, for example, *International Safeguards in Nuclear Facility Design and Construction* (IAEA, 2013a), *International Safeguards in the Design of Enrichment Plants* (IAEA, 2019a), *International Safeguards in the Design of Fuel Fabrication Plants* (IAEA, 2017a), and *International Safeguards in the Design of Reprocessing Plants* (IAEA, 2019b). Box 6.1 contains definitions relevant for nuclear nonproliferation and safeguards. Box 6.2 includes a discussion of implementation of IAEA safeguards at U.S. advanced reactor and fuel cycle facilities.

BOX 6.1
Definitions Relevant for Nuclear Nonproliferation and Safeguards

Nuclear material accountancy is "implemented by the facility operator and the State system of accounting for and control of nuclear material (SSAC) … to satisfy the requirements in the safeguards agreement between the IAEA [International Atomic Energy Agency] and the State (or group of States); and as implemented by the IAEA, inter alia, to independently verify the correctness of the nuclear material accounting information in the facility records and the reports provided by the SSAC to the IAEA" (IAEA, 2002). The IAEA considers nuclear "material accountancy as the safeguards measure of fundamental importance, with containment and surveillance as important complementary measures" (IAEA, 2011a).

> *State system of accounting for and control of nuclear material (SSAC)*: "under a comprehensive safeguards agreement, the State is required to establish and maintain a system of accounting for and control of nuclear material subject to safeguards under the agreement" (IAEA, 2002).
>
> *Nuclear material accounting*: "establishment of accounting areas, record keeping, nuclear material measurement, preparation and submission of accounting reports, and verification of the correctness of the nuclear material accounting information" (IAEA, 2002).

Estimated material conversion time is "the time required to convert different forms of nuclear material to the metallic components of a nuclear explosive device. Conversion time does not include the time required to transport diverted material to the conversion facility or to assemble the device, or any subsequent period. The diversion activity is assumed to be part of a planned sequence of actions chosen to give a high probability of success in manufacturing one or more nuclear explosive devices with minimal risk of discovery until at least one such device is manufactured. The conversion time estimates applicable at present under these assumptions are provided" in the table below (IAEA, 2002).

BOX 6.1 Continued

TABLE Estimated Material Conversion Times

Beginning Material Form	Conversion Time
Pu, HEU, or ^{233}U metal	Order of days (7–10)
PuO_2, $Pu(NO_3)_4$, or other pure Pu compounds; HEU or ^{233}U oxide or other pure U compounds; MOX or other nonirradiated pure mixtures containing Pu, U (^{233}U + ^{235}U ≥20%); Pu, HEU, and/or ^{233}U in scrap or other miscellaneous impure compounds	Order of weeks (1–3)[a]
Pu, HEU, or ^{233}U in irradiated fuel	Order of months (1–3)
U containing <20% ^{235}U and ^{233}U; Th	Order of months (3–12)

[a] This range is not determined by any single factor, but the pure Pu and U compounds will tend to be at the lower end of the range, and the mixtures and scrap at the higher end.
NOTE: HEU = highly enriched uranium; MOX = mixed oxide.
SOURCE: Adapted from IAEA (2002).

Significant quantities (SQ) of special nuclear materials indicates "the approximate amount of nuclear material for which the possibility of manufacturing a nuclear explosive device cannot be excluded." Significant quantities also take into account unavoidable losses from conversion and manufacturing processes and should not be confused with critical masses. "Significant quantities are used in establishing the quantity component of the IAEA inspection goal" (1 SQ or more of nuclear material over a material balance period), and the SQ values currently in use are given in the table below (IAEA, 2002).

TABLE Significant Quantities

Material	Significant Quantity
Direct-use nuclear material	
Pu[a]	8 kg Pu
^{233}U	8 kg ^{233}U
HEU (^{235}U ≥20%)	25 kg ^{235}U
Indirect-use nuclear material	
U (^{235}U <20%)[b]	75 kg ^{235}U (or 10 MT natural U or 20 MT depleted U)
Th	20 MT Th

[a] For Pu containing less than 80 percent ^{238}Pu.
[b] Including low-enriched, natural, and depleted uranium.
NOTE: HEU = highly enriched uranium; MT = metric ton.
SOURCE: Adapted from IAEA (2002).

Material unaccounted for represents the difference between the book inventory and the physical inventory at a facility. A nonzero material unaccounted for may indicate a diversion, but it could also be due to measurement error. In order to meet the IAEA's inspection goals, the uncertainty in determination of material unaccounted for should be small enough so that an actual diversion of 1 SQ can be detected with high probability in a timely manner (IAEA, 2002).

6.2.3 Nuclear Security

The goal of security[7] (or physical protection) is to prevent malevolent acts using radioactive or nuclear materials by substate actors. There is a distinction between the threats of radiological terrorism and nuclear terrorism. *Radiological terrorism* is an act that would lead to dispersal of radioactive materials, such as sabotage of a nuclear reactor, whereas *nuclear terrorism* is the theft of a nuclear weapon or the fissionable materials that could be used in making improvised nuclear explosive devices. In general, physical protection measures are designed to implement a strategy of "detect, delay, and respond" to prevent or mitigate a security threat. A key component of any physical protection system is the threat assessment—a determination of the magnitude and scope of the threats that a nuclear facility may face, including external physical attacks, insiders, and cyberattacks. This so-called design-basis threat (DBT) is then used to determine protective measures, including security forces, facility hardening, access controls, personnel vetting, and cybersecurity programs. The DBT is also typically graded to take into account the potential consequences of attacks, such as radiological sabotage or theft of fissionable materials with different levels of attractiveness for use in nuclear weapons.

This section describes the international and domestic (U.S.) frameworks for physical security, followed by a discussion of security measures for transporting fissionable and other radioactive materials. Transportation security is given distinct attention because the radioactive or nuclear material is mobile, outside any fixed facility protection system, and traveling on or through areas accessible to the public.

BOX 6.2
Implementation of IAEA Safeguards at U.S. Advanced Reactor and Fuel Cycle Facilities

As mentioned in Section 6.2.2, as a nuclear weapon state under the Treaty on the Non-Proliferation of Nuclear Weapons, the United States is not obligated to make all its nuclear materials and facilities available for verification by the International Atomic Energy Agency (IAEA). However, under its 1980 voluntary offer agreement with the IAEA, the United States has committed to make available for safeguards "all source or special fissionable material in all facilities within the United States, excluding only those facilities associated with activities with direct national security significance to the United States" (IAEA, 1981). The original purpose of the U.S. voluntary commitment, as stated by President Lyndon Johnson in a 1967 speech, was "to make it clear to the world that we in the United States are not asking any country to accept safeguards that we are unwilling to accept ourselves" (von Baeckmann, 1988).

To comply with this obligation, the United States maintains an Eligible Facilities List (EFL) of facilities that it offers to the IAEA for the potential application of safeguards (actually, there are two lists: one for facilities licensed by the U.S. Nuclear Regulatory Commission [U.S. NRC] and one for U.S. Department of Energy [DOE] facilities). An interagency committee called the Subgroup on IAEA Safeguards in the United States (SISUS), which is chaired by the U.S. NRC, periodically reviews the list for additions and subtractions, taking into account the national security exclusion. Any agency participating in the SISUS can invoke the national security exclusion for a facility being considered for the EFL without the need for justification.

Given resource limitations, the IAEA does not actually apply full safeguards to every facility on the EFL, but only a select few, with the objective of gaining experience in implementing new safeguards technologies, testing new safeguards equipment, and training inspectors on different facility types (Hanks, 2021). (Indeed, the United States had agreements with the IAEA in the 1960s providing opportunities for the agency to inspect a range of novel facilities.) Under a special reporting protocol that is unique to the U.S. voluntary cooperative agreement, facilities can choose to fulfill their safeguards obligations simply by providing reports to the IAEA, with no actual inspections taking place (Hanks, 2021). This reduces the

[7] The discussion of security in this report focuses on physical security. Cybersecurity, also a key aspect of the overall security of a nuclear facility, will be covered by the parallel National Academies study, Laying the Foundation for New and Advanced Reactors in the United States.

BOX 6.2 Continued

cost of meeting safeguards obligations on U.S. facilities, which are borne by the U.S. Department of State through extra-budgetary contributions to the IAEA.

The U.S. advanced reactor and fuel cycle program could provide a wealth of opportunities for the IAEA to develop and demonstrate safeguards approaches and techniques for new types of facilities and materials. This would help ensure that effective verification approaches will be available if U.S.-designed facilities are exported to non–nuclear weapon states where full-scope safeguards are required.

The IAEA has only had limited experience safeguarding fast reactors, and none at all with such designs as the Natrium reactor, which uses high-assay low-enriched uranium (HALEU)–based metallic fuel, or its associated fuel fabrication facility. Similarly, the IAEA has had little opportunity historically to demonstrate safeguards approaches at the few pebble-bed high-temperature gas-cooled reactors that have operated. Currently, the IAEA is applying safeguards to the HTR-10 test reactor and HTR-PM demonstration reactor in China, but it is believed that those activities are limited and fall short of comprehensive verification, and unlike the Xe-100, those reactors do not use HALEU (Garrett et al., 2021). Notably, molten salt–fueled reactors are completely unexplored territory for IAEA safeguards. IAEA safeguards development at pyroprocessing facilities, which may be used for solid or molten salt fuels, have been limited to certain elements of the process at small-scale facilities in the Republic of Korea, where actual fissionable material separations are not yet authorized to take place (Garrett et al., 2021).

Therefore, a commitment by the United States to place all its advanced test and demonstration reactor and fuel cycle facilities on the EFL in a timely manner, and importantly to make available full funding to the IAEA so that it will have sufficient resources to actually apply safeguards to those facilities, would be invaluable for helping to further the cause of nonproliferation. Early involvement by the IAEA in the advanced reactor and fuel cycle projects would promote the effective implementation of safeguards-by-design principles. To the extent possible, these efforts would be comprehensive and serve as models for full IAEA verification protocols in non–nuclear weapon states where advanced reactors and fuel cycle facilities may be exported. There is no indication that the U.S. government is considering such a commitment at the present time.

One potential complication is that some advanced reactors and associated fuel cycle facilities may be sited at DOE national laboratories, such as the TRISO-X (TRistructural ISOtropic [the "X" denotes the specific type of TRISO fuel used by advanced reactor company X-energy]) facility at Oak Ridge National Laboratory. The location of those facilities on-site could present difficulties for the application of IAEA safeguards if there is potential for inadvertent disclosure of information regarding activities with national security significance. However, there are numerous instances where the IAEA has been able to successfully apply safeguards at sensitive DOE sites—for example, verification of excess plutonium at the K-Area Material Storage facility at the Savannah River Site. Ease of IAEA inspector access would be a consideration in siting decisions for all new facilities.

6.2.3.1 International Physical Security Framework

The chief international instrument addressing nuclear security is the Convention on the Physical Protection of Nuclear Material (CPPNM) (IAEA, 1980) and its amendment (IAEA, 2005b). The CPPNM establishes legal obligations for protection of nuclear material used for peaceful purposes during international transport, and its amendment expanded the scope to include physical protection of nuclear facilities, nuclear material used for peaceful domestic purposes, storage, and transport. Unlike the NPT, however, there are no binding provisions for international inspections and enforcement of compliance with the CPPNM by national physical protection regimes.

The IAEA, through its Division of Nuclear Security, supports states in establishing an effective nuclear security system by assisting them with "implementation of relevant international legal instruments related to information protection; physical protection; material accounting and control; detection of and response to trafficking in such material; national response plans; and contingency measures. However, each state carries the full responsibility for nuclear security" (IAEA, 2002). To assist member states with this task, the IAEA has developed documents

and guidelines related to physical security of nuclear and radioactive materials during all phases of the nuclear fuel cycle (IAEA, 1975, 2011b, 2018b, 2021e). The CPPNM and its guidance documents address physical security for plutonium, uranium-233, and enriched uranium only, and not for radioactive materials (with the exception of spent fuel) as defined in IAEA (1980) (see also IAEA, 1975, 2011b, 2018b, 2021e). Physical security related to radioactive materials is also covered extensively by the IAEA for all operations associated with peaceful uses of nuclear energy (IAEA, 2011c, 2020a). In 2006, security publications published by the IAEA were organized under the Nuclear Security Series' four sets of publications (IAEA, n.d.-b). Member states then use this guidance as a basis for establishing their regulatory structure to ensure secure management and operations of civil nuclear activities.

Incumbent in this work is a continual assessment of the adequacy of the guidance documents with respect to the evolution of international technologies, operations, and threats. The IAEA periodically provides up-to-date guidance on all aspects associated with physical security of nuclear and radioactive materials used for peaceful purposes, as reflected in both revised and new documents, to capture current conditions (IAEA, 2015c, 2016a,b, 2019c,d, 2020a,c, 2021a,b).

This comprehensive and continuing assessment is important with respect to advanced reactors and fuel cycles. One significant cross-cutting development is the proposed use of HALEU (with uranium-235 enrichments between 10 and 20 percent) as fuel for many advanced fuel designs. Physical security implications arise for the entire operational fuel cycle supporting HALEU-fueled power reactors (e.g., fabrication, use, storage, and transportation) because they will use Category II quantities of HALEU, which will require more stringent physical protection than Category III low-enriched uranium (see Sidebar 4.1 and Table 6.1 for the definition of the security categories). Because multiple countries have expressed an interest in HALEU-fueled reactors, the IAEA will need to the address physical security implications of using this fuel, including the potential for international transport of much larger quantities of HALEU in various forms than is currently the case. For example, as discussed in Chapter 4, domestic production of HALEU sources will likely not meet demand for the foreseeable future, and HALEU may need to be transported from other countries to the United States in hexafluoride, oxide, or metallic form (Griffith, 2021; Regalbuto, 2021).

The IAEA may need to develop specific guidance on physical protection for the production and use of Category II quantities of HALEU. Although a recent IAEA report that addresses LWRs using fuel enriched to over 5 percent states that current physical security measures can be accommodated for enrichments up to 10 percent, it does not address physical security impacts for reactors using HALEU fuels enriched to 10 percent or greater (IAEA, 2020d).

Physical protection requirements for advanced reactors and fuels will become clearer as designs evolve through the demonstration and prototype stages of development. As discussed in Chapter 5, advanced reactors and fuel cycles will create unique secondary wastes not associated with spent fuel, but these wastes are not expected to need physical protection controls beyond the current guidance provided for in existing IAEA documents. In December 2018, the IAEA convened a group of external experts to review the existing physical security documents in the context of advanced reactors. In parallel, the IAEA performed an internal review of these documents. The conclusion of both reviews was that "the current Nuclear Security Guidance remains valid and sufficient to address the known concerns for the protection of advanced reactors" (Larsen, 2021). This is not to say, however, that challenges and costs will not arise in characterizing these wastes to assess how they will fit into the existing security framework.

6.2.3.2 Domestic (U.S.) Physical Security Framework

Physical security is a vital component of domestic use of the commercial nuclear fuel cycle. The U.S. Nuclear Regulatory Commission's (U.S. NRC's) regulations governing physical protection, 10 CFR 73, "Physical Protection of Plants and Materials," provides for comprehensive oversight of physical protection for special nuclear materials, as well as spent nuclear fuel. 10 CFR 73 is supported by numerous U.S. NRC regulatory guides and studies that inform the technical bases for the updated requirements for nuclear facilities and materials and provide acceptable approaches to implementing them.[8]

[8] 10 CFR 73 does not address the security of such radiological materials as radioactive sources, other than irradiated fuel. 10 CFR 37 addresses the security requirements for other radioactive materials, which are far less stringent.

TABLE 6.1 Categorization of Nuclear Materials

Material	Form	Category I	Category II	Category III[c]
Pu[a]	Unirradiated[b]	2 kg or more	Less than 2 kg but more than 500 g	500 g or less but more than 15 g
^{235}U	Unirradiated[b]			
	U enriched to 20% ^{235}U or more	5 kg or more	Less than 5 kg but more than 1 kg	1 kg or less but more than 15 g
	U enriched to 10% ^{235}U but less than 20% ^{235}U		10 kg or more	Less than 10 kg but more than 1 kg
	U enriched above natural but less than 10% ^{235}U			10 kg or more
^{233}U	Unirradiated[b]	2 kg or more	Less than 2 kg but more than 500 g	500 g or less but more than 15 g
Irradiated fuel[d]			Depleted or natural U, Th, or low-enriched fuel (less than 10% fissile content)[e,f]	

[a] All plutonium except that with isotopic concentration exceeding 80 percent in ^{238}Pu.

[b] Material not irradiated in a reactor or material irradiated in a reactor but with a radiation level equal to or less than 1 Gy/h (100 rad/h) at 1 m unshielded.

[c] Quantities not falling into Category III, as well as natural and depleted U and Th, should be protected at least in accordance with prudent management practices.

[d] The categorization of irradiated fuel in the table is based on international transport considerations. The State may assign a different category for domestic use, storage, and transport, taking all relevant factors into account.

[e] Although this level of protection is recommended, it would be open to States, upon evaluation of the specific circumstances, to assign a different category of physical protection.

[f] Other fuel which by virtue of its original fissile material content is classified as Category I or II before irradiation may be reduced one category level while the radiation level from the fuel exceeded 1 Gy/h (100 rad/h) at 1 meter unshielded.

SOURCE: Adapted from IAEA (2011b).

The U.S. NRC uses a nuclear material categorization system similar to that of the IAEA for grading the security requirements for protection against theft based on the attractiveness of the material for nuclear weapons. As shown in Table 6.1, 2 kg or more of the strategic special nuclear materials plutonium or uranium-233, or 5 kg or more of uranium-235 contained in highly enriched uranium, constitute Category I quantities of material, requiring the most stringent security measures. Security measures for Category I facilities include a requirement that licensees must protect the material from a specially designated DBT for theft that is more challenging than the DBT for radiological sabotage. The U.S. NRC does not require protection against the DBT for Category II or III facilities. There are currently only two U.S. NRC–licensed Category I facilities in the United States, where highly enriched uranium fuels are fabricated for U.S. Navy reactors, and one Category II facility, as of June 2021, the Centrus American Centrifuge Plant, where a HALEU enrichment demonstration is planned.

The U.S. NRC classification of special nuclear materials does not allow for a reduction in security requirements by crediting dilution with other materials, such as uranium-238, unless they emit high levels of external radiation; only contamination with radioactive isotopes to the extent that the material meets the "highly irradiated" standard would allow a reduction in the categorization of plutonium (see Table 6.1). However, the material classification system by the U.S. Department of Energy (DOE) does reduce the attractiveness level of plutonium or uranium-233 when diluted to 10 percent or less in a matrix that cannot be separated from the special nuclear materials with simple mechanical removal (DOE, 2019).

Since the 9/11 attacks, the U.S. NRC has engaged in comprehensive studies to assess the state of its physical protection requirements, has identified gaps, and has issued regulations and provided additional guidance to address the physical protection regime governing physical protection requirements in the United States (10 CFR

37; U.S. NRC, 2013a, 2014c). 10 CFR 37 addresses only Category 1 and 2 radioactive materials (which are outside the scope of this study),[9] and, by definition, excludes radioactive material contained in any nuclear reactor fuel assembly, subassembly, fuel rod, or fuel pellet. Such material remains governed by the physical protection requirements under 10 CFR 73.

In 2006, the U.S. NRC began an effort to develop a rulemaking for enhanced security of special nuclear material. A 2015 regulatory basis document on the draft rulemaking provides detailed background on the development of U.S. NRC physical security bases (U.S. NRC, 2015). Among other changes, the rulemaking would have introduced a graded security approach similar to that of DOE, which would have given credit for dilution of special nuclear materials to less than 10 percent and would have included updated requirements for Category II facilities.

In 2018, the U.S. NRC Commissioners directed their staff to proceed with an expedited rulemaking exclusively focused on codifying the requirements of the post-9/11 security orders for nuclear materials, and not including the other aspects that the staff had proposed. This resulted in a policy issue paper, SECY-19-0095, which gave options from the staff to the U.S. NRC Commissioners, with recommendations from the staff to discontinue the rulemaking on the basis that a mere codification of the orders was not cost-justified (U.S. NRC, 2019a). The U.S. NRC Commissioners' response in August 2021, however, disapproved the staff's recommendation to discontinue the rulemaking activity (U.S. NRC, 2021g) and instead recommended that the staff provide the Commissioners, in the form of a notation vote paper, an expanded options analysis to cover potential regulatory, resource, and timing impacts associated with the identified options.

Recently, the U.S. NRC issued an informational sheet recommending assessment of additional security measures that may be needed based on current threat assessments (U.S. NRC, 2021h). The informational sheet specifically discusses HALEU fabrication facilities as Category II facilities and the potential need for supplemental physical security measures. These facilities will still be regulated under 10 CFR 73.67 but will be evaluated on a case-by-case basis to assess adequate levels of physical security protection. Implications of increased physical protection, while unknown at this time, may be significant from a cost and operational standpoint.

Coordination between DOE and the U.S. NRC on the appropriate level of protection at domestic HALEU production facilities is vital to maintain current schedules for deployment of prototype and demonstration advanced reactors. Similarly, coordination of the two agencies with international partners is also needed in order to understand the implications of using foreign-sourced HALEU to meet DOE advanced reactor needs.

As with the IAEA guidance discussed above, secondary wastes generated from advanced reactors are not expected to require enhanced physical protection measures beyond the requirements defined in the current U.S. NRC regulations; however, there may be additional costs and challenges associated with characterizing these wastes.

6.2.3.3 Transportation Security

Transportation is the most vulnerable phase in protecting nuclear materials from a security standpoint, as the material is removed from the confines of the nuclear facility. Regulatory, technical, and operational considerations are part of an overall strategy for providing secure transport of fresh and spent nuclear fuel. This is particularly important when transporting across international boundaries, as countries have different regulations based on their own level of risk acceptance. From an international nuclear security perspective, the IAEA provides umbrella guidance of how member states can comply with their CPPNM obligations regarding physical protection of nuclear and radiological materials in transport.

Spent fuel security requirements for U.S. domestic transportation are defined in 10 CFR 73.37, which provides detailed requirements that include development of a security plan with the full range of protective components needed to properly protect such shipments during transport. This regulation references the self-protection threshold for irradiated spent fuel of a minimum radiation level of 1 Gy (gray)/hr at 1 m in an unshielded state.

For fresh fuels, current LWR designs enriched to less than 10 percent in uranium-235 falls into Category III (special nuclear material of low strategic significance). Security aspects are addressed from the graded approach

[9] Note that Arabic numerals for category numbers are assigned based on the activity content of radioactive materials, and Roman numeral category numbers are assigned based on the mass of fissionable nuclear materials. This study's scope is relevant for the latter and not the former.

and are covered in 10 CFR 73, Regulatory Guide 5.59 (U.S. NRC, 1983), and Regulatory Issue Summary 2005-22 (U.S. NRC, 2005). A security plan is required for these shipments. Requirements for physical security focus mostly on administrative aspects: advance notification, receiver confirmation, container, inspection, and responsibility for in-transit physical inspection. Shipment of HALEU source material and HALEU fresh fuel will likely fall into Category II, requiring additional physical security protection measures. As discussed in Section 6.2.3.2, physical protection of these materials will be assessed on a case-by-case basis.

The U.S. Department of Transportation (DOT) also has a role in physical security transport. 10 CFR 71.5 includes U.S. NRC citations to the DOT regulations; specifically, it references DOT regulations under 49 CFR 172(I), covering security plans. However, 49 CFR 172.804 provides for comparable security plans developed under other federal agencies.

For commercial spent nuclear fuel, the Nuclear Waste Policy Act (NWPA) (Public Law 97-425) and its Amendments (Public Law 100-203, Part E), place the authority on DOE for managing commercial spent nuclear fuel for disposal and transport from the reactor sites to the disposal facility. The NWPA further stipulates that U.S. NRC licensing shall be used to the greatest extent possible. This means that DOE will use U.S. NRC–licensed casks and other regulatory protocols, such as security requirements, as mandated by the U.S. NRC. DOE's Office of Nuclear Energy (DOE-NE) manages the program that is planning for transportation and disposal of commercial spent fuel. DOE-NE also sponsors and manages all the advanced reactor and fuel cycle programs that are being considered for the United States. Since these new concepts are for commercial power generation, DOE-NE is expected to use the U.S. NRC certification process in licensing, operation, and physical protection of these new reactors and fuel cycles.

DOE does have significant experience shipping spent fuel, nuclear waste, and nuclear materials under U.S. NRC physical security regulations. Under the Atomic Energy Act and its amendments, DOE has the authority to regulate the transport of its own materials (DOE, n.d.-a). In practice, DOE uses both self-regulation and the U.S. NRC regulations to operate its multiple energy and security programs. Within its weapons program responsibilities, DOE's National Nuclear Security Administration (NNSA) self-regulates because of the important security issues and challenges involved, coupled with the highly classified nature of much of this work. DOE has its own set of review processes to address the nature of the materials that it ships. Under DOE, the Office of Secure Transport manages the safe and secure transport of government-owned special nuclear materials. Armed federal agents provide security and incident command response in the event of an emergency (DOE-NNSA, n.d.-a).

Other examples of DOE shipment programs include naval reactor fresh and spent fuel shipments (Miles, 2016) and DOE's Office of Packaging and Transportation, located within its Office of Environmental Management (DOE-EM, n.d.). Naval reactors have dual certification for their Type B shipping containers, the M-140 and M-290 spent fuel casks.

6.2.4 Roles and Responsibilities of the U.S. Department of Energy and the National Nuclear Security Administration Relevant for This Study

The NNSA's roles and activities in nonproliferation and arms control range from securing nuclear materials and controlling dual-use technology to obstructing adversary weapons development and managing the effects of a nuclear incident. Relevant for this report are the NNSA objectives of "prevent" and "counter":

- "Prevent would-be proliferant states from developing nuclear weapons or acquiring weapons-usable nuclear material, equipment, technology, and expertise, and prevent non-state actors from acquiring nuclear and radioactive materials that can be used for malicious purposes"; and
- "Counter the efforts of both would-be proliferant states and non-state actors to acquire, develop, disseminate, deliver, or use the materials, expertise, or components of a nuclear or radiological device." (DOE-NNSA, n.d.-b)

While most of the advanced reactor and fuel cycle facilities will be regulated by the U.S. NRC, those that are owned by DOE and located at DOE and NNSA sites may not be regulated by the U.S. NRC. DOE security directives will apply to those facilities.

Box 6.3 details the roles of specific offices within NNSA related to nonproliferation, safeguards, and security that are pertinent to this study's scope.

Separately, in 2020 the DOE-NE established the Advanced Reactor Safeguards program (within the Advanced Reactor Demonstration Program), focused on assisting advanced reactor vendors with meeting domestic materials accountancy and physical protection requirements to "help reduce roadblocks in the deployment of new and advanced reactors" (Cipiti, 2021a).

However, it is also important for this DOE program to be aligned with the objectives of NNSA's Office of Defense Nuclear Nonproliferation and Office of International Nuclear Safeguards, and be informed by NNSA expertise in nonproliferation and counterterrorism, especially regarding such relevant aspects as the expanded use of HALEU and the continued development of closed fuel cycles.

6.2.5 Nuclear Material Attractiveness

An important metric for assessing the relative nuclear proliferation and nuclear terrorism risks posed by different nuclear fuel cycles is the "attractiveness" of the various nuclear materials within the fuel cycle for different adversaries seeking to acquire nuclear weapons. The *material attractiveness* of a specific type of nuclear material can be defined as its "relative utility … for an adversary in constructing a nuclear device" (Bathke, 2021). It is an intrinsic property based on the physical, chemical, nuclear, and radiological characteristics of the material. Generally, more attractive materials require more stringent extrinsic material protection, control, and accountancy measures to mitigate their proliferation and terrorism risks than do less attractive materials.

Determining material attractiveness requires knowing "the time and potential difficulties" of (1) acquiring the material; (2) processing the material into a form suitable for use in a nuclear explosive device (NED); (3) fabricating the NED; and (4) utilizing the NED (Bathke, 2021). However, since these considerations depend on

BOX 6.3
Organization and Responsibilities of the National Nuclear Security Administration (NNSA) for Nonproliferation and Safeguards

NNSA's Office of Defense Nuclear Nonproliferation (NNSA/DNN) develops and implements "policy and technical solutions to eliminate proliferation-sensitive materials and limit or prevent the spread of materials, technology, and expertise related to nuclear and radiological weapons and programs around the world" (DOE-NNSA, n.d.-b). The activities and responsibilities relevant for this report include (both on international and national level): detection and prevention of theft and smuggling of nuclear and radioactive material; prevention of sabotage of nuclear power plants and fuel cycle facilities, research reactors and nuclear material in transit; minimization of the use of highly enriched uranium (HEU) (conversion of HEU to low-enriched uranium, removal and disposal of HEU); closure of proliferation pathways (in collaboration with the International Atomic Energy Agency [IAEA]) to the acquisition of nuclear weapons, special nuclear materials, as well as relevant technology and expertise; and implementation of export control and support for negotiations and implementation of agreements and associated monitoring and verification programs.

NNSA's Office of International Nuclear Safeguards (NNSA/INS) "builds the capabilities of the IAEA and partner countries to implement safeguards obligations" (DOE-NNSA, n.d.-b). International nuclear safeguards represent various technical measures used to verify that "a country is in compliance with its legal agreements with the IAEA and not diverting nuclear material to weapons programs or pursuing undeclared nuclear activities" (DOE-NNSA, n.d.-b). NNSA/INS "meets this mission by developing and implementing new safeguards concepts, approaches, and technologies. Additionally, the program coordinates the implementation of IAEA safeguards at U.S. Department of Energy facilities and fulfills the legislative mandate to ensure that U.S.-obligated nuclear material held in foreign countries is adequately protected" (DOE-NNSA, n.d.-b).

the capabilities of the adversary, material attractiveness is not a fixed property of the material but will be different for different adversaries (Bathke, 2021). Its determination is further complicated by the fact that it also depends on the adversary's objectives.

Nuclear materials that can be used directly in a nuclear device can be attractive targets. Highly enriched uranium, plutonium (excluding nearly pure plutonium-238), and uranium-233, which are defined as "strategic special nuclear materials" by the U.S. NRC and "direct-use materials" by the IAEA, are the materials of greatest concern.[10] Although some are more readily usable than others, any nuclear material with a bare critical mass can theoretically be used in an NED and requires some level of safeguards and security protection (Bathke et al., 2012). The bare critical mass is a crucial parameter because it plays a fundamental role in determining the size and weight of a nuclear weapon, and it also sets the scale of the material acquisition effort for adversaries seeking to build a weapon.

Additional materials of concern include neptunium and americium. Although these are not defined under the IAEA statute as materials requiring safeguards, the IAEA considers them "alternative nuclear materials" and tracks them under voluntary agreements with relevant states (IAEA, 2002). In addition, other attractive isotopes, such as curium-245, are not currently designated as special or alternative nuclear materials (U.S. NRC, 2009). These materials are particularly relevant for proliferation and terrorism risks in those advanced reactor fuel cycles that separate them in quantities of significance from spent fuel, either individually or in mixtures (see Chapter 4).

In addition to the bare critical mass, other material characteristics considered important for attractiveness include the rate of decay heat generation, which affects the stability of the high explosive in a nuclear weapon, and the external dose rate, which affects the ability to safely handle, transport, and process the material.

A fourth material characteristic, the spontaneous neutron generation rate, could reduce the reliability of a nuclear explosive to achieve its design yield by increasing the likelihood of preinitiation before the necessary number of fissions occur. The degree to which these characteristics may impede their use in a nuclear explosive device will vary depending on the sophistication and technical resources available to the adversary. These parameters were incorporated by a DOE laboratory team into "figure of merit" metrics for comparing the material attractiveness of different isotopes and mixtures (Bathke et al., 2012).[11]

Other considerations that could affect the attractiveness of nuclear material include its chemical form (for instance, metal versus oxide) and the degree to which it is diluted in a non–weapons-usable matrix from which it is not readily separable by mechanical means. However, chemical means of reducing attractiveness cannot render a nuclear material entirely useless for a weapon but can only increase the time an adversary would need to convert the material into a more usable form, providing delay that would give law enforcement authorities more time to locate stolen material. This principle is the basis for NNSA's "dilute-and-dispose" approach for surplus weapons plutonium reviewed by a National Academies committee (NASEM, 2020), as well as the one of the bases of DOE's graded safeguards approach (DOE, 2019).

One key standard is the dose rate that is considered "self-protecting"—that is, a high-enough level that makes it extremely dangerous for an adversary to prepare the material for malevolent use (Coates et al., 2005). Historically, a dose rate of 100 rem (equivalent dose) (1 Sv [Sieverts]) per hour at a 1-foot distance (unshielded) was the standard for denoting "highly radioactive" materials in guidance documents (the IAEA uses 1 meter rather than 1 foot, and the U.S. NRC now uses 3 feet)—a designation that reduces the material accountability and physical protection requirements compared with special nuclear materials that are unirradiated or have dose rates below the threshold. However, this value may not be high enough to be an effective deterrent to theft (U.S. NRC, 2015). Given the need for expanded and extended spent nuclear fuel storage, reconsideration of the self-protection standard may be appropriate (U.S. NRC, 2015).

The external gamma dose rate from spent nuclear fuel is dominated by 30-year half-life cesium-137 after shorter-lived fission products decay away. Thus, the self-protecting nature of spent fuel will eventually disappear

[10] Due to a growing international consensus that the civil use of highly enriched uranium poses unacceptable security risks, such use has been nearly phased out over the past few decades.

[11] Following release of the prepublication version of this report, an erroneous sentence referring to the applicability of "figure of merit" metrics was removed.

over an extended period of time, depending on fuel burnup and other factors. The resulting consequences for the long-term security of irradiated materials in surface storage will depend on the residual attractiveness of the nuclear material after the radiation barrier falls below the self-protection threshold. A standard LWR uranium oxide spent fuel assembly with a burnup of 45 GWd/tHM (gigawatt-days per metric ton of heavy metal), which contains about 4–5 kg of plutonium (more than a Category I quantity), will remain self-protecting for well over a century after discharge using the 100 rem/hour (1 Sv/hr) standard, but for only less than 50 years using the more conservative approach based on a 1,000 rem/hour (10 Sv/hr) threshold described by Robel et al. (2013). However, even non-self-protecting uranium oxide spent fuel will be somewhat less attractive than separated plutonium because of the low plutonium concentration (approximately 1 weight percent) (Robel et al., 2013).

The committee highlights the following points concerning attractiveness of materials:

1. Nearly all isotopic mixtures of plutonium have some level of attractiveness for use in an NED; the only exception is mixtures containing 80 percent or greater of plutonium-238, which has a relatively high heat rate (NAS, 1994).
2. Given that minor actinides, such as certain neptunium, americium, and curium isotopes, may be weapon usable, reprocessing approaches such as pyroprocessing that separate plutonium in combination with one or more minor actinides do not necessarily render the product unattractive for use in an NED. The carryover of high levels of lanthanide fission products in the products of some reprocessing variants could increase the external dose rate and render the material less attractive for theft, although this would not be a significant deterrent for a proliferant state with chemical separation technology (Bari et al., 2009).
3. Thorium fuel cycles can produce attractive nuclear materials such as uranium-233. Although the admixture of uranium-232 reduces the attractiveness of separated uranium-233 through the ingrowth of the gamma-emitting decay product thallium-208, the figure of merit analysis finds that uranium-233 "is highly attractive at any practical concentration of ^{232}U" (Bathke et al., 2012).
4. The attractiveness of nuclear materials can be reduced by diluting them to a low concentration with unattractive materials. For example, isotopic dilution of uranium-235 by uranium-238 to below the highly enriched uranium threshold of 20 percent uranium-235 can only be reversed through reenrichment. In contrast, dilution with a chemically dissimilar material, such as dilution of plutonium with uranium-238, can be reversed by chemical separation, although recovery will increase the time delay before an adversary can acquire the requisite quantity of weapon-usable material (U.S. NRC, 2015).

Notably, the attractiveness of the materials throughout a given nuclear fuel cycle is one of several factors that determines its relative proliferation and theft resistance. Fuel cycles containing more attractive materials, such as those involving reprocessing, require more stringent material protection, control, and accounting measures than those with less attractive materials, such as the once-through LWR fuel cycle. This is generally referred to as "graded" safeguards and security. Techniques and approaches needed to safeguard and secure separated highly attractive materials have technical, cost, and practical limitations that might constrain their implementation, compared with the simpler measures needed for the more proliferation- and theft-resistant once-through LWR cycle. However, it is difficult to make cost comparisons because the U.S. commercial nuclear industry (as opposed to DOE) has little experience with safeguards and security measures for materials that are more attractive than low-enriched uranium for LWRs.

6.3 EVALUATION OF NONPROLIFERATION IMPLICATIONS AND SECURITY RISKS OF ADVANCED REACTORS' FUELS AND FUEL CYCLES

This section provides the committee's evaluation of nonproliferation and security risks regarding the aspects identified in the statement of task (see Sidebar 1.1 in Chapter 1) for the fuel cycles of advanced reactor types examined in this study. In particular, this section assesses HALEU, uranium-plutonium mixed oxide fuel, and advanced fuel cycles that require separating plutonium and other fissionable materials from spent fuel. The section also provides the committee's examination of nuclear material accounting and control, as well as containment,

surveillance, monitoring, and timeliness of detection related to the assessed advanced reactors' fuels and fuel cycles. The baseline reference fuel cycle is the once-through low-enriched uranium oxide cycle (as defined in Chapter 2).

In responding to the statement of task, the committee used a simplified assessment that focused on certain characteristics of advanced fuel cycle technologies that differ from the baseline uranium oxide LWR once-through cycle and could have implications for nonproliferation and security, such as types and quantities of material, as well as impacts on material accountancy and safeguards. However, a comprehensive assessment (such as the studies of the Generation IV International Forum's Proliferation Resistance and Physical Protection Working Group) was beyond the committee's scope. Such an analysis would fully consider material attractiveness and proliferation resistance features, all routes to acquisition of nuclear weapon-usable materials, and the broader technical capabilities and political context in the countries where these fuel cycles may be deployed.

6.3.1 Once-Through Fast Reactors Using HALEU

This category includes the Natrium and ARC-100 reactors (by TerraPower and ARC Clean Technology, respectively), as well as the Oklo Aurora microreactor, all descendants to varying degrees of the Experimental Breeder Reactor (EBR)-II, although with some novel features. Factors that affect their proliferation risks compared with the LWR once-through cycle are the types and quantities of nuclear material in the fresh and spent fuels and the potential diversion and misuse pathways for obtaining weapon-usable material throughout the fuel cycle. These factors also affect the security risks, with additional considerations related to the attractiveness of the nuclear materials throughout the fuel cycle, such as the physical size, chemical composition, and external radiation barriers.

6.3.1.1 Natrium (Batch-Refueled)

The Natrium reactor initially uses a metallic fuel containing an alloy of 90 weight percent uranium and 10 weight percent zirconium. TerraPower is planning a staged approach for development of the technology. The 345-MWe (megawatts electric) demonstration plant will initially use Type 1 fuel with an average enrichment of 18.5 percent HALEU and an average discharge burnup of 59,000 MWd/THM (megawatt-days/tons of heavy metal) (Hejzlar, 2021; Neider, 2021; TerraPower, 2021a). The cycle length will be 12 months—somewhat shorter than is the current practice for LWRs, which are refueled every 18 to 24 months. However, as discussed in TerraPower's 2021 presentation to the committee, by the mid-2030s, TerraPower aims to qualify and use an advanced Type 1B fuel with an average burnup of 150,000 MWd/THM and a cycle length of 18 months. In the longer term (2050 time frame), TerraPower hopes to develop even higher-burnup fuels and larger reactors that would allow reducing fuel enrichments to below 10 percent, and ultimately make possible a breed-and-burn operation, in which case the cycle length would be increased to 22 months. For the latter option (formerly known as the Traveling Wave Reactor and now called Natrium-U), proliferation risk would be expected to decrease after the reactor starts up with an initial core loading of HALEU, because subsequent fuel reloads would use only depleted uranium and thus not require uranium enrichment (Hejzlar, 2021; Neider, 2021; TerraPower, 2021a).

None of these systems are being designed as breeders, and as such do not contain natural or depleted uranium blanket fuel that would be irradiated only to low burnups and reprocessed to recover plutonium. In general, HALEU-fueled fast reactors have less favorable characteristics for breeding than plutonium-fueled reactors in any event (Kim et al., 1999).[12] In the longer term, once-through high-conversion reactors such as the Natrium-U may be able to significantly improve uranium utilization without the proliferation concerns and safeguards challenges of conventional closed-cycle fast breeder reactor systems.

The refueling frequencies for different Natrium reactor options are important to consider because, for reactors under international safeguards, inspectors must be present whenever the reactor is shut down for refueling. In this regard, the suite of Natrium reactor designs would not appear to impose an additional inspection resource burden on a per-unit basis compared with large LWRs. However, as with all smaller reactors, on a per-megawatt basis, the

[12] The Natrium-U would achieve high internal conversion by irradiating blankets to high burnups and utilizing them as driver fuel without reprocessing but would not actually be a breeder, despite often being called a breed-and-burn reactor.

overall inspection resources could be greater than those for larger reactors (IAEA, 2014). In addition, the different nature of the fuel and reactor design from that of LWRs may require changes to safeguards procedures, requiring new approaches and inspection requirements. The fresh fuel—the most attractive material on site—will contain HALEU, and therefore could require more frequent inspections if the IAEA were to decide that shorter timeliness goals were appropriate, as discussed in Section 6.3.4.5. On the other hand, Natrium (1B) and pressurized water reactor fuel assemblies each contain about the same quantity of uranium-235 (20 kg), so (unless the IAEA reduces the SQ value for HALEU) the inspection resources needed for item counting would be similar for both types of fuel.

All pool-type sodium-cooled fast reactors pose safeguards challenges. The design features a spent fuel storage area within the reactor vessel, where spent fuel discharges are first sent for cooling for up to 3 years before they are removed from the reactor vessel, cleaned of sodium, and either transferred to a water-filled spent fuel storage pool outside of the vessel or loaded into a dry canister and stored. Under LWR procedures, safeguards inspectors would be able to directly observe the refueling and verify the presence of all spent fuel declared to be in the in-vessel storage area during the required annual physical inventory. However, with sodium pool–type reactors, the fuel's location within the vessel and the opacity of the sodium limit direct visual inspection, so new approaches will be required. TerraPower told the committee that the spent fuel assemblies will be "tagged and easily identifiable," but it did not make clear how inspectors would be able to verify the tags (Hejzlar, 2021; Neider, 2021; TerraPower, 2021a). On the other hand, according to TerraPower, it will be difficult to access fuel stored in the reactor vessel, since it is a sealed system. Also, there will be a one assembly in, one assembly out restriction on the fuel handling machine, which will also have the capability to read and display the assembly tags that can be easily monitored. After the spent fuel is transferred to the water-filled pool and eventually to dry storage, the IAEA could use safeguards approaches it has experience implementing, such as the containment and surveillance measures it applied at the spent fuel pools at the Joyo and Monju sodium-cooled fast reactors in Japan, where both fresh and spent fuel assemblies were stored (Bays et al., 2021).

The security risks posed by the Natrium reactor will depend on the nuclear material content, size, weight, dilution, and other properties of fuel assemblies that would affect its vulnerability to theft. However, the fresh fuel will contain Category II quantities of HALEU and will therefore require appropriate security measures during transportation, receipt, and storage prior to irradiation.[13] Depending on the reactor design and fueling strategy, the Natrium reactor's spent fuel could contain significant plutonium and residual uranium-235. This would not initially pose a security concern because the radiation barrier will be substantial for many decades. However, the physically smaller and denser metallic fuel assemblies for the initial operation of Natrium-DEMO with sodium-bonded fuel may have lower dose rates at 1 meter than LWR spent fuels for the same burnup. Over time the increasing attractiveness of the spent fuel as the radiation barrier wanes could necessitate strengthened security measures. The higher burnup of Type 1B fuel may offset this to some extent.

6.3.1.2 Long-Life Cores Using HALEU

Both the ARC-100 and the Aurora designs use HALEU. One primary difference of these designs compared with the Natrium reactor is that they plan to use a single-batch core with a 20-year cycle length, instead of periodic refueling cycles. This will have both advantages and disadvantages for safeguards. According to the IAEA, "reduced core access and reduced refueling frequency makes misuse of the facility and diversion of spent fuel much more difficult" at reactors with sealed, long-life cores (IAEA, 2014). This advantage is also cited by advanced reactor developers (Sackett and Arthur, 2021). Since in principle the reactor vessel will not be opened over its lifetime, inspectors will not need to visit to observe refueling operations, as is the case for conventional, batch-refueled reactors. However, long-life reactor concepts "need to be reconciled with the traditional IAEA annual physical inventory of each reactor core, performed when access to the core is possible" (IAEA, 2014).

[13] The Natrium Type 1B fuel is a uranium microalloy. Each Type 1B fuel assembly would contain about 21 kg of uranium-235 in HALEU (a Category II quantity of special nuclear material), and each Type 1B fuel reload would contain about 500 kg of uranium-235 in HALEU. Thus, according to current U.S. NRC guidance, as discussed in Section 6.3.4.2, the reactor would require enhanced Category II security measures for prevention of a "gross theft" of HALEU containing 75 kg of uranium-235 (the quantity in four Type 1B fuel assemblies).

Thus, there will be no opportunity following start-up for inspectors to detect anomalies in the core inventory that may indicate a diversion or substitution. Additional containment and surveillance measures, as well as reactor monitoring, may be necessary to compensate for the reduced opportunity for direct verification.

Other factors will need to be accounted for in designing a safeguards approach for long-life cores. For example, if a long-life core reactor has the potential to experience fuel failures that would require shutdowns and fuel replacements, a reactor owner would need to plan for such contingencies either by storing additional fresh fuel on site as a backup or arranging for new fuel deliveries from off site—both activities that would require additional inspections. Also, maintenance issues may arise during plant operation that could necessitate plant shutdown and vessel access. IAEA inspector presence may be required for such off-normal events, which could present challenges for small modular reactors and microreactors being considered for deployment in remote locations.

Security considerations will also differ for the smaller, long-life core designs. Despite their small size, these reactors will require substantial quantities of HALEU to achieve criticality. For example, based on a planned burnup of 1 percent, the 1.5-MWe Aurora will require several MT (metric tons) of HALEU assemblies with enrichments of up to 19.75 percent—well over a Category II quantity. Both the ARC-100 and the Aurora will be derated significantly in order to achieve long core lives without exceeding burnup limits.

The ARC-100 reactor fuel will attain an average discharge burnup of 76,800 MWd/THM over a 20-year period—comparable to the burnup of the Natrium Type 1B fuel after 5 years (Sackett and Arthur, 2021). This discharge burnup will provide significant self-protection for the spent fuel, although it will be slower to accumulate than for Natrium fuel.

In contrast, the fuel for the first Aurora unit will only achieve a burnup of 1 percent, or less than 10,000 MWd/THM after 20 years, although Oklo hopes to later achieve 2 percent (less than 20,000 MWd/THM) and eventually exceed 60,000 MWd/MTHM (DeWitte, 2021). The concern here is that, at least for the initial units, the dose rate of the fuel will remain relatively low both during the operating cycle and at discharge, and will fall off rapidly afterward.[14]

Depending on the dose rate during irradiation, the Aurora fuel may require Category II security not only before the reactor starts operation, but also at times during operation and after shutdown. Furthermore, as discussed in Section 6.3.4.2, the reactor could require an on-site security force to ensure prompt response measures should adversaries attempt "gross theft" of HALEU—especially given plans for deployment in remote locations where off-site local law enforcement response may be slow or insufficient. The plutonium in the Aurora spent fuel may also require an enhanced level of protection. While the need for a prompt security response would not be an issue for a unit deployed at the Idaho National Laboratory site, which has its own armed response force, such a requirement may limit broader deployment and would generally conflict with the company's intention to not have an on-site security force for the Aurora and for each reactor to have facilities nearby open to the public (Oklo, 2020; Stamp, 2020).

6.3.1.3 Metallic Fuel Fabrication

Another aspect of the fuel cycle for the PRISM-type metal-fueled fast reactors relevant for safeguards is that current fuel fabrication methods generate a large quantity of unrecoverable scrap (nearly 30 percent) (Moore and Severynse, 2020). This waste stream could present challenges for material accountancy, depending on its form, how it will be stored, and the accuracy of the measurement techniques used to assay it.

The throughput of an industrial-scale fuel fabrication facility capable of supplying 1 GWe (gigawatt electric) for Natrium reactors would be on the order of 6.4 MT of HALEU per year (taking into account total scrap generation), and would therefore require Category II security, including measures to address the risk of "gross theft" of low-enriched uranium (see Section 6.3.4.2).

[14] Although the committee did not receive spent fuel dose rate information about the Aurora, a technical analysis of a conceptually similar once-through long-life (30-year) small fast reactor utilizing metallic HALEU fuel, modeled after the 10-MWe Toshiba 4S, found that the dose rate at 1 meter of spent fuel with a burnup of 34,000 MWd/THM is about 350 rem/hr at 5 years after discharge (Frieß et al., 2015). If the spent fuel is discharged halfway through the cycle after attaining a burnup of 17,000 MWd/THM, the dose rate would be 200 rem/hr after 5 years (Frieß et al., 2015). Extrapolating from these data points, the Aurora spent fuel, at a burnup of less than 10,000 MWd/THM, would likely have a dose rate 5 years after discharge not much higher than current self-protection threshold of 100 rem/hr at 1 meter.

6.3.2 Pebble-Bed Reactors Using HALEU

Pebble-bed reactors using HALEU include the Xe-100 high-temperature gas-cooled reactor (HTGR) and the Kairos fluoride-cooled high-temperature reactor. The fuel for these reactors are graphite pebbles containing TRISO fuel particles, as discussed in Chapters 3 and 4. The fuel kernels for both reactor designs consist of UCO (uranium-carbon-oxygen), with equilibrium average uranium enrichments of 15.5 percent for the Xe-100 (Mulder, 2021) and 19.55 percent for Kairos (Blandford and Peterson, 2021). Pebble-bed reactors do not necessarily require the use of HALEU, but can also use LEU+—the German Modul design, upon which the Xe-100 reactor is based, as well as the Chinese HTR-PM reactor, both use uranium dioxide fuel with enrichments below 10 percent. The use of higher enrichments, in concert with the use of UCO instead of uranium dioxide, allows for higher discharge burnups to be achieved (Mulder, 2021). However, the use of HALEU will impact both international security and domestic material accounting and security requirements. The risks will be partly offset by the large number of items needed to acquire weapons-relevant quantities of material, as well as the lack of industrial maturity of methods for reprocessing TRISO fuel.

Each 80-MWe Xe-100 reactor will require 1,540 kg of HALEU for the initial core and 430 kg per full-power year afterward, or 6.2 MT and 1.72 MT per year, respectively, for a 320-MWe 4-pack (Mulder, 2021). The Kairos Power 140 MWe reactor core will require about 525 kg of HALEU for the initial core and 600 kg per year afterward (Blandford and Peterson, 2021). On a per-unit basis, these quantities are not large in terms of the relevant low-enriched uranium SQs, which are on the order of several hundred kilograms; but as discussed below, the total material inventory at a multiunit site, including fresh and spent fuel storage, can be substantial.

Pebble-bed reactors are refueled online, which is a major defining characteristic for the safeguards approach. Online-refueled (also called "on load-refueled") reactors, such as CANDUs (Canadian deuterium uranium reactors), may require greater safeguards resources than batch-refueled reactors such as LWRs, because fuel is loaded and unloaded during operation, and not only during discrete shutdown periods when inspectors typically conduct a physical inventory.

Pebble-bed reactors present greater challenges than CANDUs for material accountancy because of the larger number of items containing fissionable material, the portability of these individual items, and the nearly continuous fueling and refueling cycles (Kovacic et al., 2021). A national laboratory team proposed that pebble-bed reactors should not be considered online refueled reactors with regard to their safeguards characteristics but should be placed in a new category of "bulk-fuel" reactors (Durst et al., 2009).[15]

This material accountancy challenge is partially offset by the large number of items that would need to be diverted to acquire a significant quantity of special nuclear material. For example, each Xe-100 fresh fuel pebble contains 7 g of HALEU, so approximately 69,000 pebbles would need to be diverted to acquire 1 SQ of low-enriched uranium (485 kg)—which is 30 percent of the core, or more than 1 year's throughput of a single Xe-100 reactor module.[16] Also, since each spent pebble would contain about 0.13 g of plutonium at peak burnup (Mulder, 2021), diversion and processing of a comparable number of pebbles—61,500—would be needed to acquire 1 SQ of plutonium. The large number of pebbles that would be have to be removed all but rules out the possibility that abrupt diversions of 1 SQ of either low-enriched uranium or plutonium from an operating module could be accomplished without detection.

Nevertheless, the potential for undetected protracted diversion or misuse cannot be excluded if the uncertainties of the material accountancy system are large enough and if all nuclear material on site is considered. In addition to each reactor module, inspectors will need to verify the inventories of fresh fuel and spent fuel storage areas. Each Xe-100 core (which contains approximately 224,000 pebbles when fully fueled) is fully replaced approximately every 3.5 years, so that a nuclear plant containing four Reactor Modules (the Xe-100 standard design to produce

[15] For comparison, a typical CANDU reactor contains 380 distinct fuel channels, with 12 fuel bundles per channel, for a total of 4,560, and loads about 17 bundles per day. In contrast, each Xe-100 module contains some 220,000 fuel pebbles and will load and circulate over 40 pebbles per hour. A model FHR that serves as the basis for the Kairos FHR will circulate 450 fuel and graphite moderator pebbles per hour, or one every eight seconds (Disser et al., 2016).

[16] Here the committee assumes that the IAEA SQ will remain 75 kg of U-235 for HALEU; if it is decreased, as discussed in Section 6.3.4.5, this would reduce the number of fresh pebbles needed to obtain 1 SQ, as well as increase the nondetection probabilities for diversions of 1 SQ.

320 MWe) will receive 10 million fresh fuel pebbles over a 40-year plant lifetime (plus replacements for damaged pebbles) and have the same number of spent fuel pebbles to store and ultimately dispose of at the end of plant life. In that context, the likelihood of timely detection of a diversion of 1 SQ may be lower than one might surmise based on the large number of items needed, and will decrease over time as spent fuel pebbles accumulate. Currently, there is no plausible way for pebbles to maintain a unique identifier over their operational life, although technical approaches have been proposed (Gitau and Charlton, 2012).

Because of the near-continuous fueling and refueling cycle, as well as other material flows in the fuel handling system, the material control and accounting system for pebble-bed reactors is envisioned as a hybrid of approaches used for reactors, which use item accounting, and those for bulk-handling facilities, such as enrichment plants, where inspectors monitor in-process nuclear material flows (Kovacic et al., 2020). Inspectors will not be able to conduct physical inventory verification of the reactor core under normal circumstances. For such reactors, the IAEA requires "dual C/S"—that is, two independent containment and surveillance (C/S) measures for each plausible diversion path (Durst et al., 2009). However, even dual C/S does not fully compensate for the inability to conduct periodic physical inventory verification, as failure of a C/S measure would cause loss of continuity of knowledge that could only be rectified by a timely physical inventory of the entire site—which may not be possible for a pebble-bed plant. Moreover, online refueled reactors require significantly greater inspector resources for application of C/S measures than batch-refueled reactors (Boyer, 2021).

For safeguards, flow monitors will be needed to count pebbles at various transfer points and to distinguish between fresh fuel, graphite moderator pebbles, and irradiated pebbles at various burnups, including spent pebbles (IAEA, 2014). The error estimates of the counting rate for current systems range from 5 percent to as low as 0.1 percent per instrument (Kovacic et al., 2020). It will be difficult to assess the effectiveness of these systems without knowing their uncertainties more precisely, and further research and development is likely required.

In addition, pebble counting alone will be insufficient to accurately determine and verify nuclear material inventories at pebble-bed reactors. The Xe-100 pebble-bed reactors use online burnup measurement systems to estimate the burnups of pebbles discharged from the core, in order to determine whether they should be reloaded into the core (e.g., partially spent) or discarded to temporary dry storage (e.g., fully spent). Such systems use gamma spectroscopy to measure concentrations of fission products such as cesium-137 as a surrogate for burnup. However, although such indirect measurements may be sufficient to determine the number of times individual pebbles have passed through the reactor, they do not accurately measure the fissionable material content of each pebble, and would be of limited use for safeguards. First, there are uncertainties in the burnup determination itself, which one study estimated as in the range of 5–10 percent (Hawari and Chen, 2005). Second, there are uncertainties of similar or greater magnitude in the depletion codes used to estimate radionuclide concentrations as a function of burnup for pebble-bed reactors, as demonstrated by a recent benchmark exercise (NEA-OECD, 2019).

Even with uncertainties on the order of 5 percent, a preliminary safeguards study of a model fluoride-cooled high-temperature reactor (FHR) similar to Kairos found that diversion of 1 SQ of low-enriched uranium from a single module would be readily detectable (Disser et al., 2016). However, the study also concluded that "the overall system non-detection probability may be called into question" for a multimodule site (12 units in the case of the model FHR) if the facility had common fresh and spent fuel storage facilities (Disser et al., 2016). The challenges of detecting a protracted diversion of one SQ would be compounded in a state with numerous multimodule sites (Gitau and Charlton, 2012).

Overall, it appears that the development of an effective safeguards approach remains a work in progress. Careful attention to integrated safeguards and security by design principles could certainly help, but fundamental challenges are likely to remain because of the inherent measurement difficulties in pebble-bed systems. However, as stated above, the large quantities of fresh or spent fuel that would have to be diverted to acquire 1 SQ are mitigating factors that would have to be considered in assessing the proliferation risks posed by pebble-bed reactors.

6.3.2.1 Proliferation-Resistance Features

Despite these challenges for material accountancy, X-energy claims that their Xe-100 HTGR is "extremely proliferation-resistant" (Mulder, 2021). X-energy told the committee that the fuel cycle renders the plutonium

isotopic mixture "useless for nuclear proliferation" because of the high in situ utilization of the plutonium-239 and -241 fissile isotopes and respective destruction/buildup of even-numbered (fertile) isotopes (e.g., plutonium-240 and -242). However, as discussed in Section 6.2.5 (Material Attractiveness), all isotopic mixtures of plutonium, except those with 80 percent or more plutonium-238, may be attractive to some degree for nuclear weapons.

A second argument is that TRISO fuel is inherently proliferation resistant because it is difficult to reprocess and there is little experience with the chemical techniques that would be required to do so, in comparison with highly mature LWR fuel reprocessing. X-energy makes the point that "TRISO particles are very tough to crack mechanically" (Mulder, 2021). The committee heard a somewhat different perspective from a BWXT representative, who said that recycling TRISO fuel is not as simple as recycling LWR fuel, though it can be done in a lab, and the challenge is to do so on an industrial scale (Lommers, 2021).

The industry is working to overcome the challenge of industrial-scale TRISO particle recycling, at least for fresh fuel. X-energy told the committee that the company will need to recover HALEU from scrap generated at its planned TRISO-X fuel fabrication plant, and accordingly, is developing methods to extract uranium from off-specification TRISO particles in order to minimize waste and to reuse the uranium (Pappano, 2021). Thus, one may expect that if a state has the capability to produce its own TRISO fuel economically, it will also have the means to extract uranium from TRISO particles—at least before irradiation. The current lack of industrial experience in reprocessing TRISO irradiated fuel particles does not imply that a determined state could not develop such capability in the future to obtain fissionable material for use in weapons.

Moreover, as discussed in Chapter 5, the large volume of contaminated graphite in spent pebble-bed fuel makes managing the waste form cumbersome and may pose problems for geologic disposal. If so, there may be an incentive to develop industrial-scale methods for separating the graphite matrix from the TRISO particles, and possibly for reprocessing the particles themselves. This would further diminish in the long-term any proliferation-resistance advantage the fuel may have today.

6.3.2.2 Security Considerations

Evaluating the security risks posed by pebble-bed reactors will require taking into account factors that affect the vulnerability of materials within the reactor and fuel cycle to theft and acquisition of sufficient material for a nuclear weapon. The primary factor is the presence of HALEU in the fresh fuel, which will require that pebble-bed reactors meet Category II security requirements, adjusted for the current threat environment as discussed in Section 6.2.3.2. In addition, the dose rate of individual pebbles is below the regulatory self-protection standard of 100 rem/hour at 1 meter and therefore may require the higher self-protection standard of 500 rem/hour value, as proposed by Chung et al. (2012).

The very small amount of nuclear material in both fresh and spent pebbles would make abrupt theft of a single Category II quantity of material challenging, not to mention a quantity sufficiently large to constitute a gross theft containing 75 kg of ^{235}U (see Section 6.3.4.2). At 7 grams of total uranium per pebble, more than 9,000 fresh pebbles would need to be stolen to obtain a Category II quantity of HALEU (64.5 kg of 15.5 percent–enriched uranium), and over seven times that number for a 485-kg "gross quantity" of HALEU. For spent fuel pebbles, which would no longer contain HALEU at the peak projected burnup but would contain 0.13 g of plutonium each (Bays et al., 2021), about 3,850 pebbles would be required for a Category II quantity of plutonium and about 15,400 for a Category I quantity.

How many transport or storage containers would an adversary need to steal to acquire these quantities of pebbles? The Versa-Pac, which is being considered for fresh fuel, is a 55-gallon drum which could hold about 350 pebbles (Kovacic et al., 2021). Over 25 Versa-Pacs would need to be stolen to obtain a Category II quantity of HALEU, and nearly 200 for a gross quantity. These quantities represent from about 4 to 14 percent of the approximately 700 containers of fresh fuel a 4-pack Xe-100 plant would require annually, making theft an unlikely scenario.

For spent fuel, the AVR-TLK container, currently in use to store fuel from the two shut-down German pebble-bed reactors, has a capacity of 950 pebbles (Bays et al., 2021). Relatively low numbers of containers would be needed to obtain a Category II quantity (5) or a Category I quantity (17) of plutonium. However, other factors

would need to be considered, including the external dose rate and overall weight of a fully loaded container. In addition, if the U.S. NRC decides to take dilution into account in revising its physical protection requirements (see Section 6.2.3.2), then the spent pebbles would likely be considered a highly diluted waste form that would require less intensive security than more concentrated items.

6.3.3 Once-Through Molten Salt–Fueled Reactors Using Low-Enriched Uranium

The material accountancy challenges discussed above for pebble-bed reactors will be even greater for molten salt–fueled reactors. While the fuel in a pebble-bed reactor may be difficult to track and count, it does consist of discrete items. In contrast, the special nuclear material in a salt-fueled reactor constantly flows through and outside of the core (Scott et al., 2021). For the purposes of safeguards, molten salt–fueled reactors can be considered bulk-handling facilities similar to reprocessing plants. This is true even for designs that are nominally once-through—that is, they do not plan to use online chemical separation facilities to extract fissionable materials or neutron-absorbing fission products from the fuel, nor to (at least initially) reprocess and recycle the spent fuel. Such reactors include the Terrestrial Energy IMSR (integral molten salt reactor) and ThorCon thermal-spectrum designs, and the fast-spectrum MCFR (molten chloride fast reactor).

Currently, it is difficult to conduct accurate and timely material accountancy at bulk-handling facilities, especially for in-process nuclear materials. The total in-process inventory cannot be measured directly during operation, but only extrapolated through such means as sampling and destructive assay, nondestructive assay, and process monitoring. The very large throughput of special nuclear material of an industrial-scale bulk-handling facility, coupled with technical limits on the accuracy and precision of measurement techniques, can lead to a large level of material unaccounted for (see Section 6.2.2), consisting of many significant quantities of special nuclear material. Even when the facilities are cleaned out and physical inventories are taken, the accumulation of difficult-to-measure residual holdup can make a significant contribution to the material unaccounted for. As a result, the IAEA cannot rely on material accountancy alone to meet its safeguards goals at bulk-handling plants and must also use containment and surveillance (C/S) and other methods.

Material accountancy at a salt-fueled reactor will be even more complex because, unlike fuel cycle facilities, the nuclear material inventory within the reactor changes with time. Because the depletion codes for salt-fueled reactors are likely to have large uncertainties, it may not be possible to precisely estimate the reactor inventory as a function of time, even if inputs and outputs are accurately measured. This poses challenges for material accountancy approaches that do not attempt to measure the nuclear material within the reactor, but rely only on C/S (Hogue et al., 2021). Moreover, it has been pointed out that it would be difficult to use C/S measures to cover the entire primary flow loop (Shoman and Higgins, 2021; Soares et al., 2020).

Consequently, a second approach that would use measurements of nuclear materials in salt-fueled reactors is being pursued, taking advantage of the systems that reactor operators would use to monitor salt composition for process control. These process monitoring approaches could include destructive and nondestructive assay of salt samples and in situ measurements of process streams (Hogue et al., 2021). Oak Ridge National Laboratory researchers are working to identify signatures that could provide indications of a diversion, as well as developing measurement techniques and instrumentation for detecting those signatures (Dion et al., 2020). However, these approaches are still immature and will require research and development under representative conditions by working with actual irradiated fuel in a reactor environment. The extreme environmental conditions within salt-fueled reactors will make "the ability to monitor and perform measurements during operation of an MSR … a severe technical challenge" (Dion et al., 2020). No research facilities are currently conducting experiments on irradiated salts in the United States, although China has recently begun to operate a small molten salt test reactor (Mallapaty, 2021; Scott et al., 2021; WNN, 2022b).[17]

A further complication is that, as for pebble-bed reactors, there are no scheduled periodic outages for refueling that would provide opportunities for safeguards inspectors to conduct physical inventory verifications. However, because of the corrosive nature of molten salts, periodic shutdowns for maintenance will likely be required, which

[17] This sentence was updated after release of a prepublication version of the report to reflect China's recent startup of this reactor.

would provide opportunities for a cleanout and inventory, although not necessarily at the frequency that inspectors would need (Hogue et al., 2021).

Establishing an effective material accountancy approach in salt-fueled reactors will be a complicated endeavor, requiring measurement of the types and quantities of nuclear materials in the fresh fuel, irradiated fuel, process stream, and waste streams. These measurement technologies have not yet been fully developed and demonstrated. An IAEA official said that molten salt-fueled reactors "raise the bar for safeguards" because there is much more of an opportunity for material to be diverted, so stronger and more elaborate extrinsic measures will be required (Boyer, 2021).

6.3.3.1 Terrestrial Energy IMSR

The Terrestrial Energy IMSR will use low-enriched uranium, with an enrichment below 5 percent; it is the only advanced non-LWR design being supported by DOE that does not use HALEU. The use of low-enriched uranium reduces the proliferation risk associated with fresh fuel production and storage relative to HALEU-fueled designs. The greater concerns are the production of plutonium in the reactor and the challenges in accounting for the material. One problematic characteristic of once-through, low-enriched uranium, salt-fueled reactors is the steady increase of the core inventory of plutonium over an operating cycle (Higgins et al., 2021). Over time, for a fixed overall measurement error, the uncertainty in material unaccounted for will increase, and the likelihood of detecting a diversion of plutonium will decrease to an unacceptable level, highlighting the need to develop improved safeguards approaches (Higgins et al., 2021) (see Figure 6.1).

6.3.3.2 ThorCon and MCFR

According to the initial design of ThorCon's proposed reactor, as presented to the committee, each 500-MWe ThorCon plant, consisting of two reactor modules, uses an average of 1.93 MT of HALEU enriched at 19.7 percent and 3.3 MT of thorium annually to generate uranium-233 in situ (Jorgensen, 2021).[18] (About one-third of the total amount of HALEU would be required for the initial core, and the remainder would be used for continuous fueling over an 8-year operating cycle.) Unlike molten salt, in uranium-233 breeders, such as that proposed by Flibe, there is no plan to separate protactinium-233 to maximize uranium-233 production, a process that would produce a separated stream of uranium-233, a direct-use material comparable to plutonium in its attractiveness for weapons. Consequently, the once-through ThorCon design poses lower-order proliferation concerns than thorium molten salt breeders. Nevertheless, the design will require a greater level of safeguards than LWRs because of the large quantity of HALEU that will be required for continuous fueling, as well as the ingrowth of both uranium-233 and, to a lesser extent, plutonium.[19] Also, although the initial design will be once-through, ThorCon might use reprocessing in the future to recover residual uranium for recycle (Jorgensen, 2021). Additional safeguards will be required in a possible future in which uranium in the spent fuel is extracted and possibly reenriched. However, this would not be done at the power plant site but rather at a separate safeguarded reprocessing site.

ThorCon currently plans to install its reactors on barges and deploy them by sea to countries such as Indonesia. For such cases of transportable nuclear power plants, early provision of design information to the IAEA is critical (IAEA, 2013b). Since each plant would presumably include sufficient fuel for its entire operating lifetime, about 15.5 MT of HALEU would have to be shipped with the plant, or more than 40 SQs (again assuming the current value of 75 kg of contained uranium-235). This substantial inventory, coupled with the storage and continuous feeding of the fresh fuel, could make the application of safeguards challenging at such a facility.

[18] In the event that HALEU is not available, the reactor can start with 4.75 percent–enriched uranium (Jorgensen, 2021a).

[19] The committee notes that during fact checking, in June 2022, ThorCon stated that its design has changed subsequent to its presentation to the committee in January 2021. Specifically, due to concerns about HALEU availability from Russia, ThorCon provided information to the committee in spring 2022 stating that it has altered its plans to use low-enriched uranium and to refuel once a year. Because of these plans, there will also be no use of thorium. The committee was not able to review the new design information after the closure of public information gathering, and the chapter's text is based on the previous information that the committee had time to review during public information gathering.

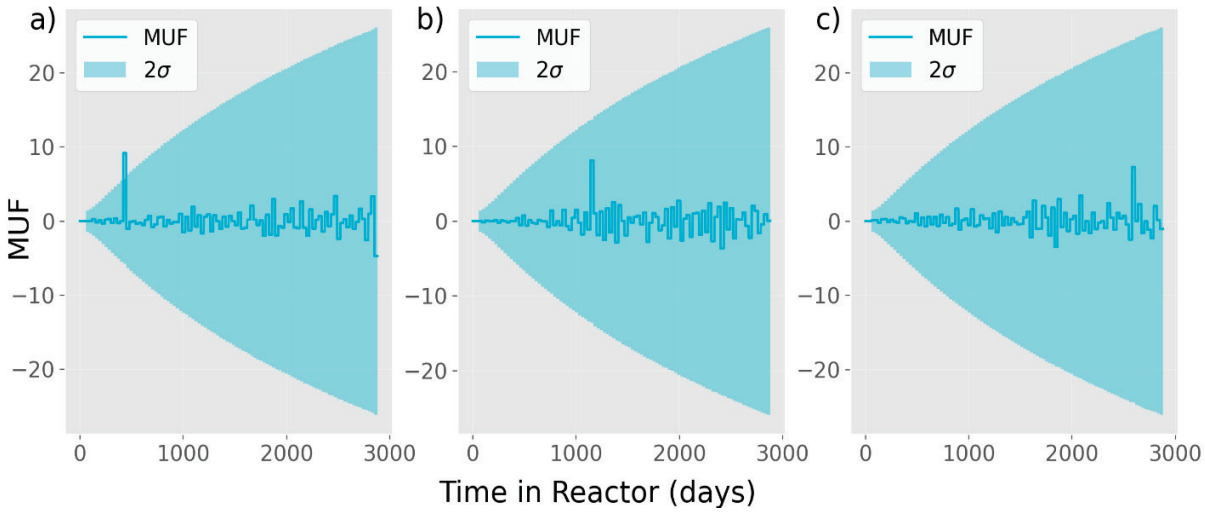


FIGURE 6.1 The average measurements of material unaccounted for (MUF) as a function of the time fuel spends in a molten-salt-fueled reactor similar to the Terrestrial Energy IMSR for the three loss scenarios: (a) early lifetime loss, (b) mid-lifetime loss, and (c) end of lifetime loss.[a]

[a] The figure caption was revised after release of a prepublication version of the report to correct information about the reactor.

SOURCE: Higgins et al. (2021). Courtesy of Sandia National Laboratories.

ThorCon's spent fuel is likely to present less of a proliferation risk than its fresh fuel. Plutonium production in its reactor will be lower than in a reactor fueled only with uranium because some uranium-238 is displaced by thorium, although this will be offset to some extent by production of uranium-233. The total quantity of uranium-233 in the spent fuel of a 500-MWe plant after 8 years would be about 400 kg, or 50 SQs, compared with about 150 kg of plutonium, or 19 SQs (Jorgensen, 2021). However, the uranium-233 would be mixed with about 10.6 MT of uranium-238, making it comparable to low-enriched uranium in its unattractiveness for weapons.[20]

The TerraPower MCFR will be initially fueled with 12 percent–enriched HALEU, and in principle, it will only require additions of depleted or natural uranium over its operating life, as its purpose is to be a high-conversion reactor (Latkowski, 2021). This would mitigate the proliferation risks to some extent compared with salt-fueled reactors requiring continuous additions of HALEU or other fissionable materials. However, the reactor would present similar challenges for material accountancy as other uranium salt-fueled reactors, since the core plutonium inventory is likely to increase to a substantial level (on the order of metric tons) as the reactor converts depleted uranium to plutonium. The committee did not receive sufficient information about the fuel cycle for this reactor to perform a detailed assessment.

6.3.4 Nonproliferation and Security Implications of the Use of HALEU

Nearly all of the advanced reactor designs currently being supported by DOE and private entities would require HALEU in the enrichment range of 13–19.75 percent, but the fuel cycle supporting commercial nuclear power production today is not designed or optimized to accommodate reactors requiring HALEU fuel. Annual HALEU needs for a single reactor could range from hundreds to thousands of kilograms, depending on the design and power

[20] There is no definition for low-enriched uranium-233 comparable with low-enriched uranium-235 in IAEA safeguards, although a proposal has been made for how to define such a quantity (Forsberg et al., 1998). However, in such a determination, the uranium-235 content would also be counted. Using the formula in Forsberg et al. (1998), the mixture would be classified as low-enriched uranium.

rating. If such reactors were to be deployed on a significant scale, the entire nuclear fuel cycle—from uranium enrichment and conversion to fuel fabrication, transport, and storage—would need to be reconfigured for the different safety and security attributes of HALEU. This is no small task. In particular, it would require security upgrades throughout the fuel cycle, because HALEU (in sufficient quantity) is classified as a security Category II material in both domestic physical protection regulations and international conventions (see Sidebar 4.1 for definitions).

Key questions include (1) whether the large-scale deployment of HALEU-fueled reactors and associated fuel cycle facilities around the world could lead to a significant increase in nuclear proliferation and nuclear terrorism risks compared with the current LWR-dominated fleet; (2) what measures would be needed to adequately mitigate those risks; and (3) what impacts would such measures have on cost, plant operations, and other factors.

It is also important to distinguish between the additional risks that the use of HALEU could pose for nuclear proliferation by states and those for nuclear terrorism by substate actors. This distinction requires understanding the extent to which access to HALEU would be more advantageous than access to lower-enriched grades of uranium for these different types of adversaries.

An adversary can use HALEU to produce an NED by three pathways: (1) direct use of HALEU in an NED; (2) further enrichment of HALEU to highly enriched uranium; and (3) use of HALEU fuel in a reactor to produce plutonium (or uranium-233, for fuel cycles using thorium-232). The latter two routes are primarily concerns for proliferation by states, since any material acquisition path that would involve further enrichment or irradiation and reprocessing is generally considered highly implausible, if not impossible, for substate actors to accomplish on their own (U.S. NRC, 2015). (However, substate groups working in concert with adversary states may have access to such capabilities.) Thus, the committee views that the only plausible option for substate actors working alone is to use stolen HALEU directly in an NED.

However, the historical motivation remains elusive for designating uranium enriched between 10 and 20 percent as Category II material, requiring more stringent physical protection than Category III uranium enriched to less than 10 percent. The initial 1975 version of the IAEA's physical protection recommendations, INFCIRC/225, introduced the designation, and it was subsequently adopted by the U.S. NRC in 1979. But the U.S. NRC's stated rationale was not concern for the direct use of the material by terrorists, but the possibility that "quantities of uranium enriched to less than 20 percent in the U-235 isotope could be diverted, without timely detection, to other countries for additional enrichment or for plutonium production" (U.S. NRC, 1978).

Thus, in 1979 the U.S. NRC presumed that low-enriched uranium would be attractive to a subnational adversary only if it were colluding with a state with enrichment or plutonium production capabilities. Consequently, the higher attractiveness of Category II enriched uranium would be due solely to the fact that less separative work would be required to enrich Category II uranium than Category III uranium to highly enriched uranium levels. However, since 1979, enrichment plants have become more compact and efficient as gas centrifuges supplanted gaseous diffusion. It is not clear that access to HALEU would still provide a major advantage relative to Category III low-enriched uranium for nations with access to modern gas centrifuge enrichment technology. Given this, with respect to the ease of further enrichment, the 10 percent lower bound would appear less important today (Boyer, 2021).

Even so, the U.S. NRC recently reaffirmed that Category II HALEU is more attractive and requires greater protection than Category III enriched uranium (U.S. NRC, 2019b, 2021f). (The complete U.S. NRC analysis is classified.)

6.3.4.1 Practicality of HALEU as a Weapons Material

Since HALEU has a uranium-235 enrichment below 20 percent and therefore is considered low-enriched uranium, the IAEA classifies it as indirect-use material, unlike the direct-use materials: highly enriched uranium, plutonium, and uranium-233. However, this classification does not mean it is impossible to use HALEU directly in an NED. In fact, any nuclear material with a finite bare critical mass can be used, in theory, to make a nuclear explosive device; however, less attractive materials present greater technical challenges (CGSR, 2000). The main question with regard to the nuclear terrorist threat posed by HALEU is whether a given quantity can be used by

a subnational group to build a sufficiently practical and deliverable NED to achieve the group's desired nuclear yield and reliability. This depends on the technical sophistication of the group in question, as well as its objectives.

In material attractiveness studies conducted by researchers at Los Alamos National Laboratory, Lawrence Livermore National Laboratory, and Pacific Northwest National Laboratory, HALEU is assigned a figure of merit between 0 and 1 and is characterized variously as "unattractive" (but "may still be theoretically possible to build a nuclear weapon or explosive device") (Bathke et al., 2012), or "impractical, but not impossible" (Ebbinghaus et al., 2013). A 2005 National Research Council study stated that "nuclear explosives can in principle be made with material containing somewhat less than 20 percent ^{235}U, but the amount of material required at enrichments below 20 percent is very large" (National Research Council, 2005). In comparison, highly enriched uranium with enrichments above 20 percent and below about 75 percent are described in the figure of merit studies as "attractive," and enrichments above about 75 percent are characterized as "preferred" (using bare critical masses calculated in Glaser [2006]).

A primary difference between uranium of different enrichments is the critical mass—that is, minimum quantity of material that can provide for a self-sustaining nuclear chain reaction for a given configuration. A solid sphere of uranium-235 metal has a "bare" critical mass (the critical mass in the absence of a neutron reflector material) of about 48 kg. For HALEU with an enrichment of 19.75 percent, the bare critical mass is about 780 kg—16 times greater than that of uranium-235 (Glaser, 2006). In a comprehensive review of nuclear proliferation in 1977, the Office of Technology Assessment (OTA) noted that a minimum enrichment of 20 percent for a "practical nuclear explosive" had been specified many years previously, and that any fissionable mixture with a bare critical mass greater than about 850 kg "could not be used to construct a nuclear explosive of any practical weight" (OTA, 1977), which would correspond to a uranium enrichment of about 17 percent (Glaser, 2006). This suggests that material in the higher HALEU enrichment range could be used in a nuclear explosive of practical weight, by the OTA standard at the time.

Little public information is available about the exact level of enrichment below which an impractical nuclear device would be precluded. The Atomic Energy Commission's "Hafstad Memorandum" of 1954 states that "the minimum enrichment which is capable of supporting a nuclear explosion with an infinite mass of material has been estimated as about 5 percent. Information from Los Alamos indicates that 10 percent enriched uranium is not suitable for any practical weapon but no definite upper limit can be set" (Brown and Glaser, 2016).

It is likely that changes in the threat environment and the advancement of technology in the decades since these statements were made have increased the cause for concern today and are driving the need for enhanced security and safeguards requirements.

6.3.4.2 The Need for Updated HALEU Security Requirements

The U.S. NRC's security regulations for Category II materials (including HALEU containing at least 10 kg of uranium-235) in 10 CFR 73.67 were originally introduced in 1979 and have not been updated. These requirements provide for a lower level of protection than the regulations for Category I facilities.

In particular, facilities possessing a Category I quantity of highly enriched uranium or other strategic special nuclear materials must be able to protect the material from theft by a DBT adversary—a violent attack by a sophisticated paramilitary group. This requirement drives the need for such facilities to have a dedicated and well-trained armed response force capable of neutralizing the DBT adversary. In contrast, the U.S. NRC does not currently require that facilities possessing Category II materials protect them from theft by a DBT adversary, nor does it generally require that members of the security organization at a Category II facility be "uniformed or armed with a firearm," nor have any "formal or comprehensive training" (U.S. NRC, 1983). However, per the U.S. NRC's Category II regulatory guidance, the agency expects the licensee must demonstrate that each security person understands the particular duties assigned to him/her and is fully qualified and trained to perform them (U.S. NRC, 1983). The function of the on-site security organization is limited to "early detection and assessment"

of security incidents, and communication (if warranted) to the U.S. NRC and local law enforcement authorities (LLEA), who would execute the response (i.e., prompt recovery of the material) (U.S. NRC, 1983).[21]

The U.S. NRC's current regulatory guidance for protecting HALEU, which dates to 1983 (U.S. NRC, 1983), identifies "gross theft" as a scenario with more serious potential consequences to public health and safety than a "minor theft" of low-enriched uranium (U.S. NRC, 1983). The U.S. NRC defines "gross theft of LEU" as "theft in a sufficiently large quantity that it could yield upon further enrichment or other processing enough material … to construct a clandestine fission explosive device" (U.S. NRC, 1983). The guidance estimates such a quantity as containing about 75 kg of uranium-235—the same value as the IAEA "significant quantity" of low-enriched uranium. According to the guidance, the U.S. NRC interprets "early detection" of a gross theft of low-enriched uranium as "detection during the attempted theft" (U.S. NRC, 1983).

The U.S. NRC has determined that these Category II security requirements need updating. One reason is because after the 9/11 attacks, the U.S. NRC required security upgrades via orders to licensed Category I and Category III facilities to address the new threat environment but was unable to do so for Category II facilities since there were none licensed at the time. Another is that the risks are "better understood than when the existing regulations were promulgated" (U.S. NRC, 2021h). Consequently, the U.S. NRC recently stated that "supplemental security measures for the protection of Category II quantities of SNM [special nuclear materials] may be required to address the current threat environment and the changing understanding of the risks associated with facilities possessing Category II quantities of SRM" (U.S. NRC, 2021h). This view is also shared by some industry stakeholders. According to a 2021 assessment of security requirements for Category II fuel cycle facilities by a former U.S. NRC official and two X-energy personnel, the current rules "do not identify appropriate or adequate security for HALEU that would be necessary today" (Rivers et al., 2021).

As discussed in Section 6.2.3.2, in 2021, the U.S. NRC commissioners asked its staff to evaluate the need for a new security rulemaking that could incorporate updated requirements for Category II materials, among other things. In the absence of a new security rulemaking, the U.S. NRC will handle applications for possession and use of HALEU at reactors and fuel cycle facilities on a case-by-case basis, and if necessary, will require additional security measures as license conditions. This process could be burdensome if the U.S. NRC receives a large number of Category II facility applications.

The U.S. NRC states that the supplemental security measures aim to change the general objective of a Category II physical protection system from early detection and assessment of external intruders to "prompt" detection and assessment. This could require enhancements in such areas as access controls, security patrols, and communication and coordination with LLEA (U.S. NRC, 2021h).

Rivers et al. (2021) go further, arguing that the physical protection system for a HALEU fuel cycle facility should be designed with sufficient delay to prevent adversaries from leaving a site with stolen HALEU, rather than simply to facilitate prompt recovery of the material after it is stolen. Their paper discusses the need to determine adversary characteristics, such as numbers and equipment—a process similar to the development of a DBT. Furthermore, the security response requirements may also take into account material characteristics, such as dilution, that could increase adversary task time. Facilities possessing at least one Category II "goal quantity," which the paper defines as 40 kg of contained uranium-235 (about one-half of the U.S. NRC's "gross quantity," corresponding to around 200 kg of 19.75 percent–enriched HALEU), would have to establish "high confidence" in a timely LLEA response. Although those authors believe that the primary responsibility for this interdiction capability could be provided by LLEA response, they conclude that "if sufficient delay cannot be incorporated into the security program to allow for timely LLEA response, it may be necessary for the site to have its own armed response force" (Rivers et al., 2021).

Another consideration under current U.S. NRC rules is that nuclear power reactors are required to have armed response forces to protect against the DBT of radiological sabotage and therefore would be able to readily comply with a similar requirement for protection against theft of HALEU, should the U.S. NRC decide to impose

[21] It is important to note that U.S. NRC guidance represents one acceptable method for meeting regulatory requirements but is not itself a requirement. However, each licensee must submit a security plan, and the guidance document serves as the U.S. NRC basis for review and audit of the plan.

one. However, this may not be the case in the future, as the U.S. NRC is currently developing a proposed rule for small modular LWRs and advanced reactors that would allow owners of such reactors to reduce or even eliminate armed response forces if they could demonstrate a radiological sabotage attack would not cause unacceptably high radiation doses to members of the public (U.S. NRC, 2018b). Such a regulatory approach could significantly reduce personnel costs for small modular reactors or microreactors—but not if they were required to maintain an armed response force to protect HALEU.

6.3.4.3 Further Enrichment of HALEU to Highly Enriched Uranium

The second route that an adversary with access to HALEU can use to acquire a nuclear weapon is further enrichment of the material to HEU levels. Less separative work is required to generate HEU from HALEU than from LEU feed material, and thus access to HALEU could be advantageous to adversaries. For example, to acquire one kilogram of uranium enriched to 90 percent (commonly referred to as "weapons-grade" highly enriched uranium), about 1750 SWU (separative work unit) would be required with 4.5 percent–enriched feed, compared with about 550 SWU with 19.75 percent–enriched feed (roughly three times smaller).[22]

Thus, access to HALEU could reduce the time or resources necessary to acquire an IAEA-defined significant quantity of highly enriched uranium. Given the three-fold reduction in SWU, a country seeking to covertly produce highly enriched uranium in a clandestine enrichment plant could obtain a given quantity in one-third the time by using HALEU feedstock rather than LWR fuel feedstock, or in the same time in an enrichment plant three times smaller. On the other hand, the difference in SWU requirements would be less significant for a state with a large commercial enrichment facility (with a capacity of millions of SWU per year) that decided to break out and overtly produce highly enriched uranium by reconfiguring the plant. In either case, the actual impact on proliferation risk would depend on specific circumstances, including the state's chosen acquisition path, its overall capabilities and technological sophistication, and its reactor and fuel cycle infrastructure.

The committee heard differing views on this matter. One expert concluded that "widespread use of HALEU could have [a] significant impact [on safeguards] at reactors and other facilities," while pointing out that the "impact will depend on the specific design of the reactor … storage plans, and refueling schedules" and how the IAEA chooses to address timeliness issues associated with the potential reenrichment of HALEU. According to this expert, while there are many aspects of advanced reactor designs that will have a greater impact on safeguards than the use of HALEU, the use of HALEU is one common factor of almost all designs being supported by the United States (Stern, 2021). However, an IAEA expert stated that HALEU would not give states "an extra edge," based on the modest difference between the SWU required to enrich 5 percent feed to highly enriched uranium compared with HALEU feed (Boyer, 2021). In the case of diversion of a single or a small number of items, such as UF_6 cylinders, to a clandestine plant, access to HALEU might reduce the time to acquire highly enriched uranium from perhaps 2 months to 1 month, which this expert did not think would significantly impact the IAEA's ability to meet its timeliness goal (Boyer, 2021).

6.3.4.4 The Current Safeguards Framework for HALEU

As discussed in Section 6.2.2, for safeguards implementation, the IAEA classifies all low-enriched uranium as "indirect-use" material with a significant quantity of 75 kg of (contained) uranium-235, and the IAEA timeliness detection goal is set at 1 year for all low enrichments. This differs from the IAEA's physical protection framework discussed in Section 6.2.3.1, in that there is no analogue in safeguards for the 10 percent enrichment threshold that distinguishes Category II from Category III enriched uranium. The committee has been unable to determine whether the reason for this disparity in categorization has a technical basis or is simply a historical relic.

Nevertheless, 1 SQ can be obtained by diverting less HALEU than LWR-grade low-enriched uranium, so

[22] This estimate assumes a tails assay of 0.25 percent in both cases. In practice, an adversary might choose a higher tails assay, depending on the amount of feedstock available and the adversary's objectives. This would reduce the SWU requirements for either type of feedstock, but would also reduce the advantage of using HALEU.

HALEU safeguards approaches will have to be somewhat more intensive to meet the IAEA's inspection goals. For example, 1 SQ of 4.5 percent–enriched uranium is 1,667 kg, whereas 1 SQ of 19.75 percent–enriched HALEU is 380 kg. The impact of these differences on safeguards would depend on specific facility and material characteristics.

HALEU production and use could also cause other safeguards issues. One expert also raised the concern that an enrichment plant producing HALEU with just under 20 percent enrichment might cause challenges for current safeguards approaches because of measurement uncertainties and the potential that small amounts of more highly enriched material could occur in process and cause false positives (Stern, 2021). Indeed, the committee heard from Centrus that such false positives could occur as the result of off-normal conditions in a centrifuge plant producing HALEU, which is why the company received approval for possession of uranium enriched up to 25 percent, although it is not authorized to draw off product greater than 20 percent (Poneman and Cutlip, 2021).

In addition, although the inspection goals do not depend on the low enrichment, the level of inspection effort actually does, through the use of a unit called the "effective kilogram" (ekg). For all uranium with an enrichment of at least 1 percent, the ekg is defined as the uranium weight multiplied by the square of the enrichment (IAEA, 2002). The number of kilograms of enriched uranium corresponding to 1 ekg decreases as the enrichment increases. One ekg is about 500 kg of 4.5 percent–enriched uranium but only about 25 kg of 19.9 percent–enriched uranium. The maximum routine inspection effort (MRIE) at a facility is a function of the quantity of enriched uranium expressed in terms of ekg (IAEA, 2002).[23] For uranium enriched to greater than 5 percent, the MRIE increases as the square root of the annual throughput (or inventory, whichever is larger) in terms of ekgs; thus, for a fixed throughput it increases linearly with the enrichment.[24] Under the current regime, the maximum routine inspection effort would be significantly greater on a per-MWd basis for the HALEU fuel facility.[25]

However, the actual routine inspection effort at low-enriched uranium fuel fabrication plants, which is determined on a facility-specific basis, has decreased significantly over time and is in recent years much smaller than the MRIE (IAEA, 1985, 2020b).[26] No public information is available regarding the relationship between enrichment and actual inspection effort. However, one expert told the committee that the IAEA may need to modify its safeguards implementation in countries possessing large quantities of HALEU (Stern, 2021).

The historical and technical basis for the ekg unit is obscure. One author describes it as "a concept introduced to establish equivalencies of different levels of enriched uranium," but does not explain what property of the uranium is being equated (Scheinman, 1987). However, the physical mass corresponding to 1 ekg, which varies inversely as the square of the enrichment E, is roughly comparable to the power-law dependence of the bare critical mass of a metal sphere as a function of enrichment (for enrichments above the approximately 6 percent threshold that have a finite bare critical mass), which is about $E^{-1.7}$ (derived from data in Glaser, 2006). So at least for enrichments above this threshold, the ekg could be regarded as a surrogate measure of the relative amount of material needed for direct use in a nuclear weapon as a function of enrichment—an interpretation that is not consistent with the IAEA's definition of all low-enriched uranium as indirect-use material.

[23] In addition, the ekg is used to establish the threshold between "facilities" (for which state parties must submit design information to the IAEA and conclude a facility attachment) and "locations outside of facilities" for the purpose of application of safeguards: a facility is "any location where nuclear material in amounts greater than one effective kilogram is customarily used." Also, states are only required to report nuclear material transfers of at least 1 ekg to the IAEA, and the IAEA only conducts more than one inspection per year at a facility or location outside a facility with a content or annual throughput of more than 5 ekg.

[24] For uranium with enrichments below 5 percent, the MRIE is determined by a different relationship.

[25] According to the formula, the MRIE for a 500-MT fuel fabrication plant fabricating 4.5 percent–enriched LWR fuel—which could fuel about 25 1,000-MWe reactors per year—would be about 50 person-days of inspection (PDI)/year. For a HALEU fuel fabrication plant providing fuel for X-energy's Xe-100 reactors, the annual throughput to supply an equivalent number of reactors would be about 150 MT per year of 15.5 percent–enriched fuel, and the MRIE for this facility would be 1,800 PDI/year.

[26] For example, according to the IAEA Safeguards Implementation Report for 2019, there were a total of 804 PDI for 21 low-enriched uranium fuel fabrication plants in states with comprehensive safeguards agreements and additional protocols in place, averaging less than 40 PDIs per facility—an order of magnitude less than the MRIE for a 500 MT/y facility (IAEA, 2020b). Most of these facilities have throughputs of at least 500 MT/y.

6.3.4.5 Is a Revision of IAEA Inspection Goals Needed for HALEU?

The IAEA has successfully implemented safeguards at many research reactors using HALEU fuel enriched to just under 20 percent around the world, but quantities at each facility are typically on the order of tens of kilograms—well below 1 SQ (Stern, 2021). In contrast, even smaller advanced reactors require many SQs of HALEU per year. Also, the throughput of fuel cycle facilities needed to produce HALEU and fabricate fuel for a fleet of such reactors could be on the order of tens or hundreds of SQs. Thus, it is important to consider whether current safeguards criteria and inspection goals need to be tightened for advanced reactor fuel cycles using HALEU. One expert told the committee that this was a fundamental issue and pointed out that, in 1977, the IAEA included in its safeguards reporting a separate category for enriched uranium comparable to security Category II, but this was not continued (Stern, 2021).

One question is whether the safeguards goals for low-enriched uranium should be changed for HALEU, which would affect both the quantity and timeliness components. This is both a technical and a diplomatic question and would likely require IAEA Board of Governors approval (Stern, 2021). As an indirect-use material, the low-enriched uranium SQ is presumably the amount needed for feed into an enrichment plant in order to produce the 25 kg of uranium-235 contained in highly enriched uranium, taking into account process losses and other difficulties with the conversion (Krass et al., 1983).

However, even if the SQ is regarded as a rough measure of the HALEU quantity needed to build a nuclear weapon without further enrichment, the current low-enriched uranium SQ of 75 kg would still make sense for HALEU. Assuming a first-generation implosion device, 1 SQ of HALEU at just under 20 percent enrichment would be about 350 kg, containing 70 kg of uranium-235.

A stronger case could be made for reducing the timeliness detection goal for HALEU from the current period of 1 year, given the shorter time needed to produce highly enriched uranium from enriching HALEU compared with lower-enriched uranium feed.

6.3.5 Proliferation and Security Risks of Fuel Cycles Using Reprocessing and Recycling

In general, closed fuel cycles using reprocessing and recycling, as discussed in Chapters 2 and 4, pose greater nuclear proliferation and security risks than once-through fuel cycles. As a result, closed fuel cycles would require the application of more intensive international safeguards and domestic physical protection measures, which can be more costly in terms of the financial, technical, and human resources needed for their implementation, than are required for once-through cycles.

In fuel cycles that involve spent fuel reprocessing, materials that are nuclear weapon usable are separated from the highly radioactive fission products that provide the self-protecting radiation barrier of spent nuclear fuel, and they are then concentrated for storage or reuse. These reprocessing products are therefore more accessible for diversion or theft than weapon-usable materials that remain bound in spent fuel—at least for the many decades after discharge that the spent fuel remains self-protecting. Depending on the fuel cycle of interest, these weapon-usable materials, as discussed in Section 6.2.5, include plutonium, 233-uranium, or transuranic elements, such as neptunium, americium, and curium. Also noted in Section 6.2.5, the residual uranium in fuels that use HALEU may be attractive as well, depending on the fuel burnup.

As discussed in Chapters 2 and 4, France and a few other countries currently operate industrial-scale reprocessing plants using the PUREX process, which allows for the separation of plutonium in a pure form. The IAEA's development of a safeguards approach for the only large-scale reprocessing plant in a non–nuclear weapon state, Rokkasho-mura in Japan (which is not yet operating), has proven to be very challenging. Because it assumes only a short time is needed for conversion of separated plutonium into a nuclear weapon, the IAEA must conduct more frequent inspections at reprocessing or mixed oxide fuel fabrication plants in order to meet its timeliness goals than it does at low-enriched uranium fuel fabrication plants or spent fuel storage facilities. In addition, the 8 kg SQ of plutonium is a small fraction of the typical annual throughput of a large reprocessing plant, which can be on the order of 8,000 kg of plutonium per year. Therefore, to meet the IAEA's quantity inspection goal, very precise and frequent in-process measurements of plutonium are required. IAEA material accountancy goals cannot

be met at large bulk-handling facilities for actinide measurement uncertainties much higher than 1–2 percent (Cipiti and Shoman, 2018). Although many techniques do exist that are below these uncertainty levels for some of the key measurements, achieving such low uncertainties throughout the process on a plant-wide basis is not feasible, necessitating implementation of a "strengthened safeguards approach" with such supplemental measures as the process monitoring used at Rokkasho (Durst et al., 2007).

The PUREX process does allow for relatively precise measurements of process streams (on the order of 0.1 percent) by sampling and destructive analysis. Of particular note is the presence of an "input accountability tank" at the head-end of the plant where dissolved spent fuel is well-mixed, sampled, and assayed, allowing precise measurement of input quantities. This step is the only way the actinide contents of the input spent fuel can be measured directly; otherwise, operators and inspectors would have to rely solely on less-accurate burnup measurements and calculations. Similarly, at the end of the process, the high purity of product materials, such as plutonium oxide, facilitates precise measurements.

However, it is impractical for safeguards inspectors to conduct sampling and destructive analysis throughout a large plant to the degree necessary to achieve such low measurement uncertainties overall. At the Rokkasho plant, which has been designed for a throughput of 800 MTHM of LWR spent fuel per year, or a plutonium throughput of about 8 MT, the IAEA established an on-site laboratory to meet its measurement requirements, which include not only intrusive monthly interim inventory verifications but also short-interval verifications requiring sampling every 7–10 days (Durst et al., 2007).

Even so, in-process nondestructive assay measurements are also necessary, but the uncertainties associated with such techniques are typically far greater than those needed to meet IAEA accountancy goals. The difficulties in meeting accountancy goals are compounded by such factors as the accumulation of residual holdup, or nuclear material in process that is difficult to recover even when the plant is cleaned out for periodic physical inventories, as well as the generation of hard-to-assay scrap and waste streams, in which substantial quantities of safeguarded materials can accumulate.

The proliferation risks of spent fuel reprocessing and plutonium recycling have long been a major concern for many experts and were the reason the Ford and Carter administrations imposed a moratorium on domestic reprocessing "unless there is sound reason to conclude that the world community can effectively overcome the associated risks of proliferation," which the Clinton administration reaffirmed in 1993 (Andrews, 2008). Nonetheless, for decades there have been numerous efforts to identify ways to modify closed fuel cycles to increase the intrinsic proliferation resistance—including the International Nuclear Fuel Cycle Evaluation and DOE's Nonproliferation Alternative Systems Assessment Program in the 1970s and 1980s. These studies, as well as many subsequent ones, have generally concluded that "there is no 'proliferation proof' nuclear power cycle" and that "all nuclear fuel cycles and many fissionable isotopes (including all those of Pu) entail some risk" (CGSR, 2000).

6.3.5.1 Proliferation-Resistance of Alternative Closed Fuel Cycles

Some experts have continued to argue that alternatives to the current PUREX-based fuel cycle that can significantly reduce proliferation and security risks (Shafer, 2021) are primarily based on reprocessing flow sheets that do not produce pure plutonium (or uranium-233) streams. As discussed in Chapter 4, such approaches include aqueous-based variations on PUREX that would coextract plutonium with uranium (COEX) or with various combinations of minor actinides, including neptunium, americium, and curium (the UREX+ suite). Some UREX+ variants would also produce mixtures containing lanthanide fission products, some of which could provide a radiation barrier. Similarly, nonaqueous technologies such as pyroprocessing, as discussed in Chapter 4, have been put forward as more proliferation resistant than PUREX because the product of normally operating pyroprocessing would be a mixture of plutonium, minor actinides, some uranium, and certain lanthanide fission products (ARPA-E, 2021).

In the 2000s, DOE initiated the Advanced Fuel Cycle Initiative (AFCI), which was intended to develop more proliferation-resistant separations processes, and later the Global Nuclear Energy Partnership, which in part sought to develop international spent fuel reprocessing and recycling centers in countries that already possessed such facilities (Andrews, 2008). The AFCI included work on demonstrating the UREX+ suite of processes and remote

recycle fuel fabrication, as well as continued pyroprocessing development. As part of the effort, DOE tasked the national laboratories with evaluating the potential benefits of such alternatives (Bari et al., 2009; Bathke et al., 2012). As discussed in Section 6.2.5, these studies concluded that none of the alternative reprocessing schemes considered conferred significant proliferation resistance compared to PUREX, because the product streams remained attractive materials for use in nuclear weapons (largely confirming what was already known) (National Research Council, 2008).

The general conclusion of these reviews was that these alternative fuel cycles would still require levels of safeguards and security comparable to PUREX-based fuel cycles.

Despite these findings, reactor developers interested in pursuing reprocessing options continue to maintain that fuel cycles that do not separate pure plutonium—most notably, pyroprocessing—do provide significant proliferation resistance. For example, Moltex, which seeks to pyroprocess spent oxide fuel to extract plutonium and other actinides for use as fresh fuel in its Stable Salt Reactor, a fast molten salt reactor, says that "the main output of the [pyroprocessing] process is an impure extraction of the minor actinides (including plutonium) … [that] would be useless in weapons" (Moltex Energy, 2022). Similarly, Flibe Energy, which is developing a thermal uranium-233 molten salt breeder reactor, ascribes great importance to the presence of uranium-232, asserting that it renders uranium-233 "highly undesirable … as a weapons material" (Flibe Energy, 2022). While these technologies may provide some benefit in delaying direct use of the materials, there was consensus among the committee members that none provided significant proliferation resistance at this time.

6.3.5.2 Impact of Alternative Fuel Cycles on Safeguards

The impact of alternative separations and fabrication technologies on the safeguardability of fuel cycle facilities and materials is another important factor in assessing their proliferation risks. As noted above, it is not possible to meet safeguards inspection goals at industrial-scale PUREX reprocessing plants and other plutonium bulk-handling facilities with material accountancy alone, even though relatively high-precision measurements can be made of the inputs and outputs. The situation will be more problematic at advanced fuel cycle facilities that do not separate pure fissionable materials and may provide fewer opportunities for precision sampling and destructive assay of process and product streams.

Aqueous reprocessing plants employing alternative fuel cycles in which plutonium is separated as part of a group of actinides and lanthanides, such as the UREX+ suite, will be able to utilize safeguards approaches similar to PUREX plants of comparable throughput. However, additional complications will arise from the fact that assays of streams containing actinide and lanthanide mixtures will be more difficult and time consuming. This will be true, in particular, for nondestructive assay, as the gamma and neutron emissions from the individual components of the mixtures, as well as their decay heat rates, can interfere with one another, obscuring the radiation or thermal signatures used to identify and measure different isotopes. Even destructive assay would likely require additional preparation and measurement time for more complex mixtures.

6.3.5.3 Material Accountancy at Pyroprocessing Plants

For nonaqueous recycle systems such as pyroprocessing, the difficulties associated with measuring heterogeneous material streams are compounded by additional technical differences between aqueous reprocessing plants that further complicate material accountancy. A recent article by Hoyt et al. (2021) notes,

> Material accountancy in electrochemical fuel reprocessing is challenging due to complex flows of material, which include continuous phase change processes, batch transfers of material of unknown composition, and multiple salt recycle streams. Additionally, elevated temperatures and corrosive fluids preclude the use of standard process monitoring technology.

More specifically, Cipiti et al. (2021a) state the following:

- Completely flushing out an electrochemical facility is not as feasible an operation as at an aqueous facility. If the operator decides not to flushout material, inventory measurements can be used instead. However, although holdup measurements may be no more difficult with pyroprocessing than any bulk-handling facility, they will be of greater importance if the plant is not periodically flushed out.
- The input spent fuel and the in-process electrorefiner salt are heterogeneous, which introduces large sampling errors.
- The uranium and transuranic metallic products can be measured using sampling and destructive analysis; however, this measurement is different from aqueous techniques in regard to the need to sample a molten metal, and "may be difficult and costly for routine measurements" (Cipiti et al., 2021a). "More experimental work is necessary" to develop techniques for accurate nondestructive assay measurement of the actinide composition of the metallic uranium/transuranic product (Cipiti et al., 2021a).

The impact of these additional challenges is apparent from the results of the DOE study known as the "MPACT" (Materials Protection, Accounting, and Control Technologies) 2020 Milestone. This study entailed development of a Virtual Facility Distributed Test Bed for a model electrochemical reprocessing (pyroprocessing) plant to demonstrate the implementation of safeguards and security by design (Cipiti et al., 2021a,b). The throughput of the model plant is 100 MT/y of LWR spent fuel, or about 1 MT of plutonium per year. This is one-eighth the size of a large aqueous reprocessing plant such as Rokkasho and would only be able to supply enough plutonium (and other transuranic elements) to fuel about one PRISM-sized fast burner reactor[27] per year.

The study estimated the detection probabilities for a range of material-loss scenarios, evaluating both the IAEA's safeguards goals and the U.S. NRC's MC&A requirements for Category I facilities (see Tables 6.2 and 6.3).[28] For the IAEA, the study used the goal of detecting the diversion of 1 SQ of plutonium (8 kg) within one material balance period (here taken as 1 month) with a 95 percent probability of detection and a 5 percent false alarm rate.[29] The study also considered the U.S. NRC's more stringent domestic material control and accounting requirements in 10 CFR 74.53(b) for detecting losses of one Category I quantity of strategic special nuclear material (2 kg for plutonium) from accessible process areas within 7 days, with a 95 percent probability.[30]

The MPACT study evaluated several loss scenarios while varying the assumed total measurement uncertainty (from both random and systematic errors) from 1 to 5 percent at the critical measurement points (input fuel, electrorefiner salt inventory, and product) (Cipiti et al., 2021a). These results indicate that the IAEA's goal can only be met for all scenarios if the total uncertainty is kept to 1 percent or less for the critical measurement points, and that the U.S. NRC's 7-day goal (for Category IB material) cannot be met except for abrupt diversions assuming the lowest measurement uncertainty category.

The actual performance of the material accountancy system would likely be lower than indicated by these results, which are based on optimistic assumptions. For example, they assume short and likely impractical material balance periods—8 days for the U.S. NRC and 30 days for the IAEA. The impacts of holdup were not considered in detail, even though holdup has been found to be significant in actual pyroprocessing operations (Vaden, 2007). Waste streams were assumed to have very low actinide content, and high measurement uncertainties were tolerated, even though experience has shown that some waste streams, such as cladding hulls, still contain a significant percentage of undissolved spent fuel (about 5 percent of the initial uranium) (Westphal et al., 2013). Sampling

[27] PRISM (Power Reactor Innovative Small Module) is a small modular, sodium-cooled fast reactor developed by GE-Hitachi Nuclear Energy, with proposed power output of 840 MWth/311 MWe (Triplett et al., 2012).

[28] These requirements would not apply to a U.S. NRC–licensed reprocessing facility today because its regulations declare reprocessing facilities to be Category II facilities by fiat, even though they would possess Category I quantities of material. This regulatory inconsistency would have to be addressed if the U.S. NRC were to receive a reprocessing plant application in the future.

[29] This is only a subset of the IAEA's Safeguards Criteria, which generally require that abrupt diversions of 8 kg of Pu be detected within 1 month and protracted diversions within 1 year (Johnson, 2009). The MPACT project's use of "abrupt" and "protracted" here needs clarification.

[30] This is only one of several U.S. NRC loss-detection goal requirements for Category I facilities. The U.S. NRC also requires that loss of a formula quantity Category IA materials (generally strategic special nuclear material items that can be carried by one person inconspicuously) has to be detected within 3 days with 95 percent likelihood. Some items, such as uranium/transuranium ingots, would likely be classified at Category IA.

TABLE 6.2 Safeguards Modeling Results Based on IAEA Detection Goal (8 kg Pu in 1 month)

	Detection Probabilities and SEID as a Function of Measurement Uncertainty (RSD)		
Loss Scenario	All 1%	All 3%	All 5%
Abrupt Loss	100%	99%	63%
Protracted Loss 1	100%	93%	31%
Protracted Loss 2	100%	66%	13%
SEID (kg Pu)	1.9	5.5	9.1

NOTE: IAEA = International Atomic Energy Agency; RSD = relative standard deviation; SEID = standard error of the inventory difference.
SOURCE: Adapted from Cipiti (2021b).

TABLE 6.3 Safeguards Modeling Results Based on U.S. NRC Detection Goal (2 kg Pu in 7 days)

	Measurement Uncertainty (RSD)		
Loss Scenario	All 1%	All 3%	All 5%
Abrupt Loss	97%	14%	7%
Protracted Loss	83%	7%	5%
SEID (kg Pu)	1.2	3.0	4.9

NOTE: RSD = relative standard deviation; SEID = standard error of the inventory difference; U.S. NRC = U.S. Nuclear Regulatory Commission.
SOURCE: Adapted from Cipiti (2021b).

errors were not specifically evaluated, even though those have been recognized as major sources of uncertainty both for the input spent fuel and for the electrorefiner salt.[31] In particular, the salt itself is heterogeneous, as it forms several different layers with dispersed insoluble particles (Croce et al., 2021). And finally, an industrial-scale pyroprocessing facility would likely have a throughput several times larger than the plant modeled in MPACT—making it even harder to detect small diversions.

Given the numerous difficulties identified above that a pyroprocessing plant poses for material accountancy, the MPACT project concluded that achievement of 1 percent measurement uncertainties is a "best-case" assumption and in most cases is "an extrapolation of experimental work and represent[s] a best estimate of what may be possible with the technology" (Cipiti et al., 2021a). A recent survey of candidate measurement techniques reveals that most are in an early stage of development, and none have yet been validated under fully representative conditions, given the absence of a production-scale facility where they could be field tested (Coble et al., 2020). The MPACT campaign's Virtual Facility Distributed Test Bed included experimental work at the DOE laboratories to further develop pyroprocessing measurement approaches that could help overcome the main technical limitations, including microcalorimetry, high-dose-rate neutron detectors, and cyclic voltammetry (see Box 6.4).

6.3.5.4 Material Accountancy at Molten Salt–Fueled Reactors with Reprocessing

As discussed in Chapter 4, some molten salt–fueled reactor concepts include online reprocessing systems to separate actinides and neutron-absorbing fission products from the irradiated fuel. Such systems would enable such designs to take full advantage of the flexibility in fuel composition and potentially very high burnup that liquid fuels could provide. They would also be needed to increase natural resource utilization compared with once-through molten salt reactors. The most prominent example is the thorium/uranium-233 thermal breeder, of which Flibe is an example. In order to maximize breeding performance, protactinium-233 must be separated and allowed to decay to uranium-233 outside of the reactor to prevent parasitic absorption and production of uranium-234, and other neutron poisons must be removed to maximize the neutron economy. This requires frequent passes of the

[31] The MPACT study simply assumed that sampling errors were included in the overall uncertainty estimates.

BOX 6.4
Proposed Approaches for Pyroprocessing Material Accountancy

The following describes three approaches—microcalorimetry, high-dose-rate neutron detectors, and cyclic voltammetry—being developed for pyroprocessing material accountancy.

Microcalorimetry is a nondestructive assay technique that holds promise for improving the precision of measuring plutonium and other actinides in the input and output streams. A microcalorimeter measures the temperature change from a single photon emission using sensors at the edge of the superconducting-normal transition, allowing for precise energy determinations. This has the potential to improve the resolution of gamma peaks by an order of magnitude compared with high-purity germanium detectors, reportedly allowing plutonium isotopic ratios to be measured to within 1 percent precision with count times on the order of hours, which may be sufficient as a substitute for destructive analysis (Croce et al., 2021).[a] In principle, this method could be applied at all the critical measurement points of the process. The results, however, would have to be coupled with additional measurements, such as traditional calorimetry, or burnup code calculations, to obtain the overall isotopic concentrations and thus the mass of total plutonium (as well as those of the other actinides), which would introduce additional uncertainties.

Work is also under way to develop passive neutron detectors that can function in high-gamma-dose-rate environments. Conventional neutron detectors used for nondestructive assay of plutonium and other actinides are sensitive to gamma-ray backgrounds, which limits their utility for measurements of impure mixtures at pyroprocessing plants, such as the uranium/transuranic product. Los Alamos National Laboratory has developed a High-Dose Neutron Detector that utilizes boron-lined proportional counters with argon and CO_2 gas, which reportedly can function with gamma dose rates up to 800 rem/hour with a 30 percent reduction in neutron detection efficiency, allowing for measurements of the neutron multiplication to less than 5 percent uncertainty (Croce et al., 2021). However, even under low gamma-background conditions, neutron multiplicity counters have fairly high uncertainties in determining plutonium mass, especially when the sample composition is not well characterized and may contain light-element impurities. As is the case for aqueous technologies, the presence of curium isotopes in the mixture, with spontaneous neutron emissions four orders of magnitude greater than the even-numbered plutonium isotopes, limits the usefulness of neutron detection for quantitative nondestructive assay. Coble et al. (2020) assumes a 10 percent overall uncertainty for neutron multiplicity counters in electrochemical plant applications.

To address the likely inability to conduct frequent plant flushouts and direct inventory measurements of the electrorefiner salt, cyclic voltammetry is also being studied as a method for measuring in-process actinide concentrations. This technique applies a varying potential to the electrorefiner salt and measures the induced current. The magnitudes of peaks in the current occurring at known redox potentials are then used to derive the concentrations of the corresponding actinide species (Coble et al., 2020).

Cyclic voltammetry has been shown to obtain precise results (<1 percent uncertainty) for uranium and plutonium chloride concentrations up to about 2 weight percent (Hoyt et al., 2021; Tylka et al., 2015). The technique, however, becomes more difficult to apply at higher concentrations more representative of production-scale facilities, where UCl_3 concentrations can approach 10 weight percent. Moreover, concentration measurements must be combined with bulk salt mass measurements to obtain actinide weights, introducing additional uncertainty, and inhomogeneities in the salt must be taken into account. Thus the MPACT best-case assumption of one percent total uncertainty for actinide mass measurements in the electrorefiner salt seems quite optimistic at this time. Another study estimates an uncertainty of 10 percent for the technique (Coble et al., 2020). Such high uncertainties would call into question a safeguards approach that relies on in-process nondestructive assay measurements for calculating the material balance.

[a] This is the statistical uncertainty; systematic errors such as sampling errors were not incorporated (Croce et al., 2021).

core through the reprocessing system. Fast salt-fueled reactors are somewhat more tolerant of neutron poisons than thermal reactors but still require their separation at a reduced rate.

The addition of online reprocessing systems would clearly increase the difficulty of achieving accurate material accountancy at molten salt–fueled reactors, compounding the already formidable challenges described above for once-through molten salt reactors due to the inability to accurately estimate and verify the core inventory. The presence of a salt processing system would provide more opportunities for diversion, thereby increasing the credibility of scenarios that take advantage of the uncertainties in accountancy measurement to conceal diversions. This is of particular concern in breeder systems that require separation of relatively pure streams of weapon-usable material, such as uranium-233.

Although the ingrowth of uranium-232 due to the decay of protactinium-232 is often cited as a proliferation resistance measure because of the intense gamma emissions of its decay products, the presence of uranium-232 is also a hindrance because it could interfere with material accountancy measurements if it requires items to be shielded (Evans-Worrall, 2021). Moreover, the uranium-232 contamination level of uranium-233 can be minimized by separating uranium-232 from protactinium-233 before protactinium-233 decays to uranium-233, taking advantage of the significantly shorter half-life (1.7 days) of protactinium-232, which might be desirable from an occupational safety perspective.

Given the chemical forms of their fuels, nonaqueous separations technologies would be needed for the online reprocessing systems at molten salt reactors. Chloride salt–based systems could use pyroprocessing technologies similar to the one discussed in Section 6.3.5.3, and the material accountancy challenges discussed there would generally apply to an on-site reprocessing unit at a molten salt reactor. Some aspects, such as the inability to accurately account for input into the processing line, would increase the difficulty. Designs that require quasicontinuous processing would introduce additional complications compared with the batch process analyzed in the MPACT study. And the steadily increasing in-core inventories for some systems illustrated in Section 6.3.3.1 would effectively increase the fissionable material throughput in the system, and hence increase the challenges of detecting small diversions over time.

For fluoride salts, analogous pyrochemical processes could be used (Fredrickson et al., 2018). Although the material accountancy aspects of such systems have not been studied in as much detail as the MPACT project's study of chloride-based pyroprocessing, one could expect similar challenges. One additional problem presented in systems containing uranium-233 is that techniques for uranium-233 nondestructive assay do not currently exist (Evans-Worrall, 2021).

6.3.5.5 Security Aspects of Alternative Closed Fuel Cycles

The question of whether alternative closed fuel cycles can be designed to have significantly lower security risks than PUREX-based systems depends fundamentally on whether the materials that are produced, stored, processed, and transported are of low attractiveness for substate actors seeking to acquire nuclear weapons. This depends on characteristics such as their vulnerability to theft, their direct weapon usability, and the ease of converting those materials to more attractive materials. Depending on these features, alternative fuel cycle materials might have greater theft resistance even though they may not be more resistant to proliferation. The 2009 Brookhaven National Laboratory study of proliferation risk reduction concluded that alternative fuel cycles producing mixtures of plutonium, neptunium, americium, curium, and possibly lanthanide fission products, would provide "some advantage" over fuel cycles that produced separated plutonium or plutonium and neptunium because of "larger radiation and heat loads" that "would be more difficult to handle for health and safety reasons" (Bari et al., 2009).

Although most lanthanide fission products are relatively short-lived, they are the largest contributors to radiation dose and heat rates of the alternative fuel cycle mixtures. But they must be removed before fabrication of current-technology fast reactor fuels because of their propensity to migrate to the fuel periphery and contribute to fuel-cladding chemical interaction (FCCI). In contrast, molten salt fast reactors may be able to retain them, as FCCI is obviously not an issue.

The 2009 Brookhaven study also concluded that "even with the lanthanides present the total dose is not very

high and would be unlikely to deter an adversary who was willing to accept injury or self-sacrifice" (Bari et al., 2009). It also pointed out that all such mixtures would still require Category I security as defined by the U.S. NRC and IAEA (Bari et al., 2009).

Nevertheless, in more refined security categorizations, such as the DOE-graded safeguards table or the comparable approach that underlies the enhanced security rulemaking that the U.S. NRC may once again consider (see Section 6.2.3.2), factors such as significant dilution by uranium or high-dose-rate and decay-heat materials might be credited in reducing attractiveness, and hence might warrant somewhat less stringent security measures than would separated plutonium. However, generally high rates of dilution (<10 percent) or a high concentration of lanthanides (>10 weight percent) are required to reduce the attractiveness of the product by one level (Bathke et al., 2012). Thus, processes such as COEX (50-50 uranium-plutonium mixture) or pyroprocessing do not significantly reduce attractiveness.

If resumed, the U.S. NRC rulemaking on enhanced security would provide an opportunity to clarify the attractiveness of all materials that may be present in advanced reactor fuel cycles and the appropriate levels of security to apply to them. A comprehensive rule would include appropriate treatment of weapon-usable materials, such as neptunium, that the current framework does not address, consideration of physical and chemical properties of materials, and a reevaluation of the dose-rate threshold for effective self-protection.[32]

[32] In SRM-SECY-09-0123, the U.S. NRC commissioners disapproved a staff proposal to include americium and neptunium in a revised material categorization scheme. Given the potential that advanced reactor fuel cycles involving separated streams or mixtures that include these elements may be developed in the future, the U.S. NRC should revisit this decision in its consideration of resuming the enhanced security rulemaking.

References

10 CFR 20(E). "Radiological Criteria for License Termination," U.S. Nuclear Regulatory Commission. https://www.nrc.gov/reading-rm/doc-collections/cfr/part020/part020-1401.html.

10 CFR 37. "Physical Protection of Category I and Category II Quantities of Radioactive Material," U.S. Nuclear Regulatory Commission, https://www.nrc.gov/reading-rm/doc-collections/cfr/part037/index.html.

10 CFR 50. "Domestic Licensing of Production and Utilization Facilities," U.S. Nuclear Regulatory Commission. https://www.nrc.gov/reading-rm/doc-collections/cfr/part050/index.html.

10 CFR 50.2. "Domestic Licensing of Production and Utilization Facilities: Definitions," U.S. Nuclear Regulatory Commission. https://www.nrc.gov/reading-rm/doc-collections/cfr/part050/part050-0002.html.

10 CFR 51.23. "Environmental impacts of continued storage of spent nuclear fuel beyond the licensed life for operation of a reactor." U.S. Nuclear Regulatory Commission. https://www.nrc.gov/reading-rm/doc-collections/cfr/part051/full-text.html#part051-0023.

10 CFR 71. "Packaging and Transportation of Radioactive Material," U.S. Nuclear Regulatory Commission. https://www.nrc.gov/reading-rm/doc-collections/cfr/part071/index.html.

10 CFR 71.5. "Transportation of Licensed Material," U.S. Nuclear Regulatory Commission, https://www.nrc.gov/reading-rm/doc-collections/cfr/part071/part071-0005.html.

10 CFR 71.55. "General Requirements for Fissile Material Packages," U.S. Nuclear Regulatory Commission. https://www.nrc.gov/reading-rm/doc-collections/cfr/part071/part071-0055.html.

10 CFR 72. "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste," U.S. Nuclear Regulatory Commission. https://www.nrc.gov/reading-rm/doc-collections/cfr/part072/full-text.html.

10 CFR 73. "Physical Protection of Plants and Materials," U.S. Nuclear Regulatory Commission, https://www.nrc.gov/reading-rm/doc-collections/cfr/part073/full-text.html.

10 CFR 74.53. "Process Monitoring," U.S. Nuclear Regulatory Commission. https://www.nrc.gov/reading-rm/doc-collections/cfr/part074/full-text.html#part074-0053.

40 CFR 190. "Environmental Radiation Protection Standards for Nuclear Power Operations," U.S. Environmental Protection Agency. https://www.ecfr.gov/current/title-40/chapter-I/subchapter-F/part-190.

49 CFR 172, Subpart I, "Safety and Security Plans," U.S. Department of Transportation. https://www.ecfr.gov/current/title-49/subtitle-B/chapter-I/subchapter-C/part-172/subpart-I.

49 CFR 173. "Shippers—General Requirements for Shipments and Packagings," U.S. Department of Transportation. https://www.ecfr.gov/current/title-49/subtitle-B/chapter-I/subchapter-C/part-173.

49 CFR 173.420, "Uranium hexafluoride (fissile, fissile excepted, and non-fissile)," U.S. Department of Transportation. https://www.ecfr.gov/current/title-49/subtitle-B/chapter-I/subchapter-C/part-173#173.420.

Adams, R. 2021. China's high temperature reactor—pebble bed modular (HTR-PM) achieves first criticality. Atomic Insights (September 14). https://atomicinsights.com/chinas-high-temperature-reactor-pebble-bed-modular-htr-pm-achieves-its-first-criticality.

Ainger, J. 2022. "EU Lawmakers Remove Last Hurdle to Label Gas, Nuclear as Green." Bloomberg. July 6, 2022. https://www.bloomberg.com/news/articles/2022-07-06/eu-lawmakers-remove-last-hurdle-for-gas-nuclear-as-green.

AIP (American Institute of Physics). 2021. "FY22 Budget Outlook: DOE Applied Energy R&D." FYI Science Policy News 78(August 26). https://www.aip.org/fyi/2021/fy22-budget-outlook-doe-applied-energy-rdd.

Aitkaliyeva, A., M. Tonks, J. Hirschhorn, J. Powers, I. Greenquist, and B. Beeler. 2020. "Research Needs for Uranium-Zirconium- Based Metallic Fuels." Idaho Falls, ID: Idaho National Laboratory, June 1. https://doi.org/10.2172/1634988.

Ancius, D., V. Remeikis, A. Plukis, R. Plukiene, D. Ridikas, and M. Cometto. 2005. "Evaluation of the Activity of Irradiated Graphite in the Ignalina Nuclear Power Plant RBMK-1500 Reactor." Nukleonika 50(3):113-120.

Andrews, A. 2008. "Nuclear Fuel Reprocessing: U.S. Policy Development." Washington, DC: Congressional Research Service. March 27. https://sgp.fas.org/crs/nuke/RS22542.pdf.

ANDRA (Agence nationale pour la gestion des déchets radioactifs). 2006. "Inventaire National Des Déchets Radioactifs et Des Matières Valorisables." Paris, France: Agence nationale pour la gestion des déchets radioactifs. https://inventaire.andra.fr/sites/default/files/documents/pdf/fr/in_-_edition_2006_-_rapport_de_synthese.pdf.

ANS. 2020. "BWXT Restarts TRISO Fuel Manufacturing." NuclearNewswire, November 12. https://www.ans.org/news/article-2369/bwxt-restarts-triso-fuel-manufacturing.

Apted, M., J. Kessler, M. Kozak, M. Nutt, A. Orrell, A. Sowder, and P. Swift. 2012. "'Radiotoxicity Index': An Inappropriate Discriminator for Advanced Fuel Cycle Technology Selection." In WM2012 Conference. Phoenix, AZ. https://www.osti.gov/servlets/purl/1078789.

Arcos, D., F. Grandia, C. Domènech, A. M. Fernández, M. V. Villar, A. Muurinen, T. Carlsson, P. Sellin, and P. Hernán. 2008. "Long-Term Geochemical Evolution of the near Field Repository: Insights from Reactive Transport Modelling and Experimental Evidences." Radiogeochemical Aspects of Nuclear Waste Disposal 102(3):196-209 (December 12). https://doi.org/10.1016/j.jconhyd.2008.09.021.

Arm, S. T., D. E. Holcomb, R. L. Howard, and B. Riley. 2020. "Status of Fast Spectrum Molten Salt Reactor Waste Management Practice." PNNL-30739 (December). Pacific Northwest National Laboratory. https://www.pnnl.gov/main/publications/external/technical_reports/PNNL-30739.pdf.

Arostegui, D. A., and M. Holt. 2019. "Advanced Nuclear Reactors: Technology Overview and Current Issues." Congressional Research Service. R45706. April 18. https://www.everycrsreport.com/files/20190418_R45706_86fb03d4ca6ab0e3f37bb71cfe23f44274a0ce84.pdf.

ARPA-E (Advanced Research Projects Agency–Energy). 2021. "Optimizing Nuclear Waste and Advanced Reactor Disposal Systems (ONWARDS)," Funding Opportunity Announcement, May 19, 2021.

Back, C., and R. Schleicher. 2021. "General Atomics Electromagnetic Systems." Presentation to the Committee on Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Reactors, January 13, virtual meeting. https://www.nationalacademies.org/event/01-11-2021/merits-and-viability-of-different-nuclear-fuel-cycles-and-technology-options-and-the-waste-aspects-of-advanced-nuclear-january-11-2021-and-january-13-2021-meeting.

Balakrishna, P. 2012. "ThO_2 and $(U,Th)O_2$ processing—A review," Natural Science 4(Special Issue):943-949. http://dx.doi.org/10.4236/ns.2012.431123.

Ball, S. J., and S. E. Fisher. 2008. "Next Generation Nuclear Plant Phenomena Identification and Ranking Tables (PIRTs)." Prepared for U.S. Nuclear Regulatory Commission. Oak Ridge, TN: Oak Ridge National Laboratory. https://www.nrc.gov/reading-rm/doc-collections/nuregs/contract/cr6944/v1/index.html.

Barber, G. 2022. "The Nuclear Reactors of the Future Have a Russia Problem." Wired. March 17, 2022. https://www.wired.com/story/the-nuclear-reactors-of-the-future-have-a-russia-problem.

Bari, R., L-Y Cheng, J. Phillips, J. Pilat, G. Rochau, I. Therios, R. Wigeland, E. Wonder, and M. Zentner. 2009. "Proliferation Risk Reduction Study of Alternative Spent Fuel Processing." Brookhaven National Laboratory. BNL-90264-2009-CP (July). https://www.bnl.gov/isd/documents/70289.pdf.

Baron, P., S. M. Cornet, E. D. Collins, G. DeAngelis, G. Del Cul, Yu. Fedorov, J. P. Glatz, V. Ignatiev, T. Inoue, A. Khaperskaya, I. T. Kim, M. Kormilitsyn, T. Koyama, J. D. Law, H. S. Lee, K. Minato, Y. Morita, J. Uhlíř, D. Warin, and R. J. Taylor. 2019. "A Review of Separation Processes Proposed for Advanced Fuel Cycles Based on Technology Readiness Level Assessments." Progress in Nuclear Energy 117(November):103091. https://doi.org/10.1016/j.pnucene.2019.103091.

Barrasso, J. 2022. "Barrasso Leads Bill to Ban Russian Uranium Imports." John Barrasso: United States Senator—Wyoming. March 17, 2022. https://www.barrasso.senate.gov/public/index.cfm/2022/3/barrasso-leads-bill-to-ban-russian-uranium-imports.

Bates, B. 2020. "HALEU Production at the Savannah River Site (SRS)." Presented at the HALEU Webinar, April 28, 2020. https://gain.inl.gov/HALEU_Webinar_Presentations/09-Bates,SRNL_Capabilities-28Apr2020.pdf.

Bathke, C. G. 2021. "Attractiveness of Materials in Advanced Nuclear Fuel Cycles." LA-UR-21-25980. Presentation to the Committee on Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Reactors, July 20, virtual meeting. https://www.nationalacademies.org/event/07-20-2021/merits-and-viability-of-different-nuclear-fuel-cycles-and-technology-options-and-the-waste-aspects-of-advanced-nuclear-reactors-meeting-8-july-20-2021.

Bathke, C. G., B. B. Ebbinghaus, B. A. Collins, B. W. Sleaford, K. R. Hase, M. Robel, R. K. Wallace, K. S. Bradley, J. R. Ireland, G. D. Jarvinen, M. W. Johnson, A. W. Prichard, and B. W. Smith. 2012. "The Attractiveness of Nuclear Materials in Advanced Nuclear Fuel Cycles for Various Proliferation and Theft Scenarios." Nuclear Technology 179(1):5-30. https://doi.org/10.13182/NT10-203.

Bays, S. E., G. A. Reyes, M. J. Schanfein, R. H. Stewart, and N. P. Martin. 2021. "Significant Quantity Production Rates in Small Modular Reactors." INL/CON-21-63833-Revision 0 (August). Idaho National Laboratory.

Bělovský, L. 2019. "The ALLEGRO experimental gas-cooled fast reactor project." Presentation to the Generation IV International Forum, March 20, webinar. https://www.gen-4.org/gif/jcms/c_105738/ladislav-belovsky-webinar-annoucement-final.

Berger, M. O. 2018. "The Nuclear Option: Technology to Extract Uranium From the Sea Advances." The New Humanitarian, June 28, 2018. https://deeply.thenewhumanitarian.org/oceans/articles/2018/06/28/the-nuclear-option-technology-to-extract-uranium-from-the-sea-advances.

Bespala, E. V., M. V. Antonenko, D. O. Chubreev, A. V. Leonov, I. Yu Novoselov, A. P. Pavlenko, and V. N. Kotov. 2019. "Electrochemical Treatment of Irradiated Nuclear Graphite." Journal of Nuclear Materials 526:151759. https://doi.org/10.1016/j.jnucmat.2019.151759.

Betti, M., L. Aldave de las Heras, A. Janssens, E. Henrich, G. Hunter, M. Gerchikov, M. Dutton, A. W. van Weers, S. Nielsen, J. Simmonds, A. Bexon, T. Sazykina. 2004. "Results of the European Commission Marina II Study Part II—Effects of Discharges of Naturally Occurring Radioactive Material." Papers from the International Conference on Radioactivity in the Environment, Monaco, 1-5 September 2002 74(1):255-277. January 1. https://doi.org/10.1016/j.jenvrad.2004.01.021.

Bisplinghoff, B., M. Lochny, J. Fachinger, and H. Bruecher. 2000. "Radiochemical Characterization of Graphite from Juelich Experimental Reactor (AVR)." Nuclear Energy (1978) 39(5):311-315.

Bixler, N. E., F. Gelbard, D. L. Y. Louis, and J. Phillips. 2017. "Review of Spent Fuel Reprocessing and Associated Accident Phenomena." NUREG/CR-7232. Albuquerque, NM: Sandia National Laboratory, February. https://www.nrc.gov/docs/ML1704/ML17045A577.pdf.

Black, G., and S. Peterson. 2018. "Economic Impact Report: Construction and Operation of a Small Modular Reactor Electric Power Generation Facility at the Idaho National Laboratory Site, Butte County, Idaho." Regional Economic Development Organization for East Idaho. https://www.uidaho.edu/-/media/UIdaho-Responsive/Files/president/direct-reports/mcclure-center/Research-Reports/redi-economic-impact-report-pdf.pdf.

Black, G. A., F. Aydogan, and C. L. Koerner. 2019. "Economic Viability of Light Water Small Modular Nuclear Reactors: General Methodology and Vendor Data." Renewable and Sustainable Energy Reviews 103(April):248-258. https://doi.org/10.1016/j.rser.2018.12.041.

Blandford, E., and P. Peterson. 2021. "Kairos Power." Presentation to the Committee on Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Reactors. January 13, virtual meeting. https://www.nationalacademies.org/event/01-11-2021/merits-and-viability-of-different-nuclear-fuel-cycles-and-technology-options-and-the-waste-aspects-of-advanced-nuclear-january-11-2021-and-january-13-2021-meeting.

Boyer, B. 2021. "Proliferation Resistance Using Methodology of the International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO)." Presentation to the Committee on Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Nuclear Reactors, September 14, virtual meeting. https://www.nationalacademies.org/event/09-13-2021/merits-and-viability-of-different-nuclear-fuel-cycles-and-technology-options-and-the-waste-aspects-of-advanced-nuclear-reactors-meeting-9-september-13-15-2021.

BRC (Blue Ribbon Commission on America's Nuclear Future). 2012. Report to the Secretary of Energy. January. U.S. Department of Energy. https://www.energy.gov/sites/prod/files/2013/04/f0/brc_finalreport_jan2012.pdf.

Breeder Reactor Corporation. 1985. Final Report: The Clinch River Breeder Reactor Plant Project. l98 160205 001. Breeder Reactor Corporation. https://www.nrc.gov/docs/ML1806/ML18064A893.pdf.

Brown, A., and A. Glaser. 2016. "On the Origins and Significance of the Limit Demarcating Low-Enriched Uranium from Highly Enriched Uranium." Science and Global Security 24(2):131-137. https://doi.org/10.1080/08929882.2016.1184533.

Brown, N., Worrall, A., and Todosow, M. 2017. "Impact of Thermal Spectrum Small Modular Reactors on Performance of Once-through Nuclear Fuel Cycles with Low-Enriched Uranium." Annals of Nuclear Energy 101(March):166-173. https://doi.org/10.1016/j.anucene.2016.11.003.

Bunn, M. 2021. "The Economics of Reprocessing and Recycling vs. Direct Disposal of Spent Nuclear Fuel." Presented at the National Academies' Committee on Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Nuclear Reactors, online, June 7. https://www.nationalacademies.org/event/06-07-2021/merits-and-viability-of-different-nuclear-fuel-cycles-and-technology-options-and-the-waste-aspects-of-advanced-nuclear-reactors-meeting-6-part-2-june-7-2021.

Bunn, M., Fetter, S., Holdren, J., and van der Zwaan, B. 2003. "The Economics of Reprocessing vs. Direct Disposal of Spent Nuclear Fuel." DE-FG26-99FT4028. Project on Managing the Atom. Cambridge, MA: Harvard University. https://scholar.harvard.edu/files/matthew_bunn/files/bunn_the_economics_of_reprocessing_vs_direct_disposal_of_spent_nuclear_fuel.pdf.

Bushuev, A. V., V. N. Zubarev, and I.M. Proshin. 2002. "Composition and Content of Impurities in Graphite of Industrial Reactors" Radiation Protection Dosimetry 92(January 1):298-302.

Caponiti, A. 2020a. "Status of Advanced Reactor Demonstration Programs." Presentation at the Gateway for Accelerated Innovation in Nuclear (GAIN) Workshop on HALEU, April 28, webinar. https://gain.inl.gov/HALEU_Webinar_Presentations/03-Caponiti,ARDP_Status_for_HALEU_Workshop,Rev1-28Apr2020.pdf.

Caponiti, A. 2020b. "Support for Advanced Reactor Development and Deployment." Presentation to the Committee on Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Reactors, September 22, virtual meeting. https://www.nationalacademies.org/event/09-21-2020/merits-and-viability-of-different-nuclear-fuel-cycles-and-technology-options-and-the-waste-aspects-of-advanced-nuclear-reactors-september-21-23-2020-meeting.

Caponiti, A. 2022. Communication from Alice Caponiti, Deputy Assistant Secretary for Reactor Fleet and Advanced Reactor Deployment in the Office of Nuclear Energy, February 20, 2022.

Carbonnier, J.-L. 2006. "Merits of Fast Reactors for an efficient use of uranium ore and reduction of ultimate waste." MIT Symposium: Rethinking the Nuclear Fuel Cycle, Cambridge, MA. October 30-31.

Carmack, W. J., S. L. Hayes, J. M. Harp, R. S. Fielding, S. A. Maloy, and T. A. Saleh. 2017. "Overview of the U.S. DOE Fast Reactor Fuel Development Program." In International Conference on Fast Reactors and Related Fuel Cycles. Yekaterinburg, Russian Federation. https://media.superevent.com/documents/20170620/2c79371c8ad3f55aad0a98836c2a24fb/fr17-207.pdf.

Castelli, R., T. Gervais, D. Favet, and B. Ytournel. 2009. "COEX™ Pellet Fabrications in MELOX Test Chain." In Proceedings of GLOBAL 2009. Paris, France, 2009.

CBO (Congressional Budget Office). 2007. "Costs of Reprocessing Versus Directly Disposing of Spent Fuel." Testimony of Peter R. Orszag, Director, before the Senate Committee on Energy and Natural Resources. Washington, DC: Congressional Budget Office. November 14, 2007. https://www.cbo.gov/publication/19318.

CEA (Alternative Energies and Atomic Energy Commission). 2008. "Treatment and Recycling of Spent Nuclear Fuel: Actinide Partitioning—Application to Waste Management." A Nuclear Energy Division Monograph. France: Alternative Energies and Atomic Energy Commission, 2008.

CEA. 2009. "Nuclear Waste Conditioning." A Nuclear Energy Division Monograph. France: Alternative Energies and Atomic Energy Commission, 2009. https://www.cea.fr/english/Documents/scientific-and-economic-publications/nuclear-energy-monographs/CEA_Monograph5_Nuclear-waste-conditioning_2009_GB.pdf.

CEA. 2015. "Advances in Research on Partitioning and Transmutation and Plutonium Multi-Recycling in Fast Neutron Reactors," Nuclear Energy Division, June 2015. https://www.cea.fr/english/Documents/corporate-publications/advances-research-on-partitioning-transmutation-and-plutonium-multi-recycling-in-fast-neutron-reactors.pdf.

CEA. n.d. "About the JHR." Jules Horowitz Reactor. https://jhrreactor.com/en/about.

CGSR (Center for Global Security Research). 2000. "Proliferation-Resistant Nuclear Power Systems: A Workshop on New Ideas, June 2-4, 1999." UCRL-JC-137954/CGSR-2000-001. Livermore, CA: Lawrence Livermore National Laboratory, March 2000. https://cgsr.llnl.gov/content/assets/docs/CGSR_2000_001.pdf.

Chiba, S., T. Wakabayashi, Y. Tachi, N. Takaki, A. Terashima, S. Okumura, and T. Yoshida. 2017. "Method to Reduce Long-Lived Fission Products by Nuclear Transmutations with Fast Spectrum Reactors." Scientific Reports 7(1):13961. October 24. https://doi.org/10.1038/s41598-017-14319-7.

Choi, E.-Y., and S. M. Jeong. 2015. "Electrochemical Processing of Spent Nuclear Fuels: An Overview of Oxide Reduction in Pyroprocessing Technology." Progress in Natural Science: Materials International 25(6):572-582. December 1. https://doi.org/10.1016/j.pnsc.2015.11.001.

Chung, K., D. Beddingfield, W. H.Geist, S.-Y. H. Lee, 2012. "Development of a Reliable Fuel Depletion Methodology for the HTR-10 Spent Fuel Analysis" LA-UR-12-22447. https://permalink.lanl.gov/object/tr?what=info:lanl-repo/lareport/LA-UR-12-22447.

Cipiti, B. 2021a. "Advanced Reactor Safeguards Program Overview." Presented at the GAIN Advanced Reactors Safeguards and Security Workshop, April 2021. https://gain.inl.gov/SiteAssets/2021-April_SafeguardsAndSecurityWorkshop/Presentations/Safe_Secure_Day1_Website/01-ARS%20Overview-Cipiti.pdf.

Cipiti, B. 2021b. "Safeguards and Security Analysis for Fuel Cycle Facilities." Presentation to the Committee on Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Nuclear Reactors, May 17, virtual meeting. https://www.nationalacademies.org/event/05-17-2021/merits-and-viability-of-different-nuclear-fuel-cycles-and-technology-options-and-the-waste-aspects-of-advanced-nuclear-reactors-may-17-19-2021-meeting.

Cipiti, B. B., and N. Shoman. 2018. "Bulk Handling Facility Modeling and Simulation for Safeguards Analysis." Science and Technology of Nuclear Installations 2018 (October 4):3967621. https://doi.org/10.1155/2018/3967621.

Cipiti, B. B., M. Browne, and M. Reim. 2021a. "The MPACT 2020 Milestone: Safeguards and Security by Design of Future Nuclear Fuel Cycle Facilities." Journal of Nuclear Materials Management 49(1):5-21.

Cipiti, B. B., N. Shoman, and P. Honnold. 2021b. "Safeguards Modeling for Advanced Nuclear Facility Design." Journal of Nuclear Materials Management 49(1):35-47.

Coates, C.W., B. L. Broadhead, A. M. Krichinsky, R. W. Leggett, M. B. Emmett, and J. B. Hines. 2005. "Radiation Effects on Personnel Performance Capability and a Summary of Dose Levels for Spent Research Reactor Fuels." ORNL/TM-2005/261. Oak Ridge, TN: Oak Ridge National Laboratory, December 2005. https://technicalreports.ornl.gov/cppr/y2007/rpt/124368.pdf.

Coble, J. B., S. E. Skutnik, S. N. Gilliam, and M. P. Cooper. 2020. "Review of Candidate Techniques for Material Accountancy Measurements in Electrochemical Separations Facilities." Nuclear Technology 206(12):1803-1826. December 1. https://doi.org/10.1080/00295450.2020.1724728.

Collins, E. D., G. D. Del Cul, B. B. Spencer, R. T. Jubin, C. Maher, I.-T. Kim, H. Lee, Y. S. Fedorov, V. F. Saprykin, V. I. Beznosyuk, A. B. Kolyadin, P. Baron, M. Miguirditchian, C. Sorel, Y. Morita, R. Taylor, A. Khaperskaya, C. Hill, R. Malmbeck, J. Law, G. de Angelis, L. Boucher, X. Xeres, E. Mendes, T. Inoue, J. P. Glatz, M. Kormilitsyn, J. Uhlíř, V. Ignatiev, J. Serp, S. Delpech, K. Ikeda, S.-I. Koyama, M. Kurata, Y. Morita, K. Tsujimoto, and K. Minato. 2014. "Technology Readiness Assessment of Partitioning and Transmutation in Japan and Issues Toward Closed Fuel Cycle." Progress in Nuclear Energy 74:242-263.

Crawford, D. C. 2020. "VTR Fuel Cycle and Waste Management." Presented at the National Academies' Committee on Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Nuclear Reactors, online, December 7. https://www.nationalacademies.org/event/12-07-2020/merits-and-viability-of-different-nuclear-fuel-cycles-and-technology-options-and-the-waste-aspects-of-advanced-nuclear-december-7-8-2020-meeting.

Crawford, D. C., C. E. Baily, C. R. Clark, R. S. Fielding, and S. C. Marschman. 2019. "Fuel Fabrication Facility Study for FCF HALEU." Idaho Falls, ID: Idaho National Laboratory, March 2019. https://gain.inl.gov/MicroreactorProgramTechnicalReports/Document-INL-EXT-19-52614_Rev.0.pdf.

Crimi, F P. 1995. "Shippingport Station Decommissioning Project ALARA Program." In Proceedings of the Third International Workshop on the Implementation of ALARA at Nuclear Power Plants. Hauppauge, Long Island, NY: Brookhaven National Laboratory. https://www.osti.gov/biblio/54675.

Croce, M., D. Henzlova, H. Menlove, D. Becker, and J. Ullom. 2021. "Electrochemical Safeguards Measurement Technology Development at LANL." Journal of Nuclear Materials Management 49(1):116-134.

Croff, A. G., and S. L. Krahn. 2016. "Comparative Assessment of Thorium Fuel Cycle Radiotoxicity." Nuclear Technology 194(2):271-280. May 1. https://doi.org/10.13182/NT15-46.

Croff, A. G., R. G. Wymer, L. L. Tavlarides, J. H. Flack, and H. G. Larson. 2008. "Background, Status, and Issues Related to the Regulation of Advanced Spent Nuclear Fuel Recycle Facilities." NUREG-1909. ACNW&M White Paper. Washington, DC: U.S. Nuclear Regulatory Commission, June 2008. https://www.nrc.gov/docs/ML0821/ML082100043.pdf.

Dalton, D. 2022. "Canda/Waste Organisation Unveils Plans for Spent Nuclear Fuel Transport." Nucnet. January 7, 2022. https://www.nucnet.org/news/waste-organisation-unveils-plans-for-spent-nuclear-fuel-transport-1-5-2022.

Deep Isolation. 2002. "Technology." Deep Isolation. https://www.deepisolation.com/technology. Accessed February 16, 2022.

Del Cul, G. D., B. B. Spencer, C. W. Forsberg, E. D. Collins, and W. S. Rickman. 2002. "TRISO-Coated Fuel Processing to Support High-Temperature Gas-Cooled Reactors." Oak Ridge, TN: Oak Ridge National Laboratory, September 2002. https://info.ornl.gov/sites/publications/Files/Pub57144.pdf.

Delpech, S., E. Merle-Lucotte, D. Heuer, M. Allibert, V. Ghetta, C. Le-Brun, X. Doligez, and G. Picard. 2009. "Reactor Physic and Reprocessing Scheme for Innovative Molten Salt Reactor System." Fluorine and Nuclear Energy 130(1):11-17. January 1. https://doi.org/10.1016/j.jfluchem.2008.07.009.

Demkowicz, P. A. 2019. "TRISO Fuel: Design, Manufacturing, and Performance." INL/MIS-19-52869-Revision-0. Idaho Falls, ID: Idaho National Laboratory, July 2019. https://inldigitallibrary.inl.gov/sites/sti/sti/Sort_24838.pdf.

Demkowicz, P. A., B. Liu, and J. D. Hunn. 2019. "Coated Particle Fuel: Historical Perspectives and Current Progress." Journal of Nuclear Materials 515:434-450.

DeWitte, J. 2021. "Oklo." Presentation to the Committee on Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Reactors, February 23, virtual meeting. https://www.nationalacademies.org/event/02-22-2021/merits-and-viability-of-different-nuclear-fuel-cycles-and-technology-options-and-the-waste-aspects-of-advanced-nuclear-reactors-february-22-and-23-2021-meeting.

Dion, M. P., L. G. Worrall, S. Croft, and L. M. Scott. 2020. "Molten Salt Reactor Signatures and Modeling Study." ORNL/SPR-2020/1836, November 2020. Oak Ridge National Laboratory.

Disser, J., E. Arthur, and J. Lambert, 2016. "Preliminary Safeguards Assessment for the Pebble-Bed Fluoride High-Temperature Reactor (PB-FHR) Concept." INL/CON-16-38642. Idaho Falls, ID: Idaho National Laboratory. September 2016.

DNFSB (Defense Nuclear Facilities Safety Board). 2021. "Defense Nuclear Facilities Safety Board Staff Report: Savannah River Site H-Canyon Safety Basis Review." https://www.dnfsb.gov/sites/default/files/document/24261/H-Canyon%20Exhaust%20Tunnel%20at%20the%20Savannah%20River%20Site%20%5B2021-100-030%5D.pdf.

DOE (U.S. Department of Energy). 1999. "Surplus Plutonium Disposition Final Environmental Impact Statement." DOE/EIS-0283. Washington, DC: U.S. Department of Energy, November 1999. https://www.energy.gov/sites/prod/files/EIS-0283-FEIS_Vol3b-1999.pdf.

DOE. 2000. "Record of Decision for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel." Federal Register 65(182). https://www.govinfo.gov/content/pkg/FR-2000-09-19/pdf/00-24005.pdf.

DOE. 2008. "U.S. Department of Energy Releases Revised Total System Life Cycle Cost Estimate and Fee Adequacy Report for Yucca Mountain Project." Energy.gov, August 5, 2008. https://www.energy.gov/articles/us-department-energy-releases-revised-total-system-life-cycle-cost-estimate-and-fee.

DOE. 2009a. Federal Register, Notices, 74(123):31017-31018. June 29, 2009.

DOE. 2009b. Research Needs for Fusion-Fission Hybrid Systems. Report of the Research Needs Workshop (ReNeW). Gaithersburg, MD: U.S. Department of Energy, October 30, 2009. https://www.osti.gov/servlets/purl/972501-uKmLlN.

DOE. 2013. "Strategy for the Management and Disposal of Used Nuclear Fuel and High-Level Radioactive Waste." Washington, DC: U.S. Department of Energy, January 2013. https://www.energy.gov/sites/prod/files/Strategy%20for%20the%20Management%20and%20Disposal%20of%20Used%20Nuclear%20Fuel%20and%20High%20Level%20Radioactive%20Waste.pdf.

DOE. 2015a. "Advancing Clean Electric Power Technologies." In Quadrennial Technology Review: An Assessment of Energy Technologies and Research Opportunities, 2015. https://www.energy.gov/sites/prod/files/2017/03/f34/quadrennial-technology-review-2015_1.pdf.

DOE. 2015b. Tritium and Enriched Uranium Management Plan Through 2060: Report to Congress. October 2015, http://fissilematerials.org/library/doe15b.pdf.

DOE. 2016. "Notice of Intent to Prepare a Supplemental Environmental Impact Statement for Disposition of Depleted Uranium Oxide Conversion Product Generated from DOE's Inventory of Depleted Uranium Hexafluoride." Federal Register 81(166):58921-58923.

DOE. 2017. "Alternatives for the Disposal of Greater-Than-Class C Low-Level Radioactive Waste and Greater-Than-Class C-Like Waste." Report to the U.S. Congress. Washington, DC: U.S. Department of Energy, November 2017. https://www.energy.gov/sites/prod/files/2018/09/f55/GTCC-2017-Report-to-Congress-on-Disposal-Alternatives.pdf.

DOE. 2019. "DOE Standard: Nuclear Material Control and Accountability." DOE-STD-1194-2019. Washington, DC: U.S. Department of Energy, September 2019. https://www.standards.doe.gov/standards-documents/1100/1194-astd-2019/@@images/file.

DOE. 2021. "Department of Energy: Nuclear Energy Advisory Committee." Federal Register 86(240). December 17, 2021. https://www.govinfo.gov/content/pkg/FR-2021-12-17/pdf/FR-2021-12-17.pdf.

DOE. 2022a. "DOE Awards $36 Million to Reduce Waste from Advanced Nuclear Reactors." Energy.gov, March 10, 2022. https://www.energy.gov/articles/doe-awards-36-million-reduce-waste-advanced-nuclear-reactors.

DOE. 2022b. "Energy Department Requests Proposals to Build and Test Second High-Tech Cask Railcar Design." Energy.gov, January 24, 2022. https://www.energy.gov/ne/articles/energy-department-requests-proposals-build-and-test-second-high-tech-cask-railcar.

DOE. 2022c. "DOE/EIS-0542: Final Environmental Impact Statement." Energy.Gov. May 13, 2022. https://www.energy.gov/nepa/articles/doeeis-0542-final-environmental-impact-statement.

DOE. 2022d. "High Assay Low Enriched Uranium (HALEU) Cascade Demo Completion and Extended Production." SAM.gov. 2022. https://sam.gov/opp/c9400feb96f04a988d142731f60ededc/view.
DOE. 2022e. "Spent Nuclear Fuel Management, Accelerated Basin De-Inventory Mission for H-Canyon, at the Savannah River Site" Federal Register 87(76): 23504–23506. https://www.govinfo.gov/content/pkg/FR-2022-04-20/pdf/2022-08383.pdf.
DOE. n.d.-a. "List of Licensed Transportation Packaging." https://www.energy.gov/em/packaging-and-transportation.
DOE. n.d.-b. "WIPP Site." U.S. Department of Energy's Waste Isolation Pilot Plant. https://wipp.energy.gov/wipp-site.asp. Accessed August 12, 2021.
DOE-EM (Office of Environmental Management). 1997. "Linking Legacies: Connecting the Cold War Nuclear Weapons Production Processes to Their Environmental Consequences." U.S. Department of Energy, January 1997. https://www.energy.gov/sites/default/files/2014/03/f8/Linking_Legacies.pdf.
DOE-EM. 2019. "Nuclear Regulatory Commission's Low-Level Radioactive Waste Classifications." https://www.energy.gov/sites/default/files/2019/06/f63/Nuclear-Regulatory-Commission%E2%80%99s-Low-Level-Radioactive-Waste-Classifications-June-2019.pdf.
DOE-EM. 2021. "Fact Sheet: Overview of the Moab UMTRA Project," July 2021. https://www.gjem.energy.gov/documents/factsheets/OverviewFactSheet_July2021.pdf.
DOE-EM. n.d. "Packaging and Transportation." Energy.gov. https://www.energy.gov/em/packaging-and-transportation.
DOE-NE (Office of Nuclear Energy). 2002. "The History of Nuclear Energy." Washington, DC. https://www.energy.gov/sites/prod/files/The%20History%20of%20Nuclear%20Energy_0.pdf.
DOE-NE. 2007. "Global Nuclear Energy Partnership Strategic Plan." GNEP-167312, Rev. 0. Washington, DC: U.S. Department of Energy, January 2007.
DOE-NE. 2019. "Advanced Reactor Technology Development Fact Sheet." Energy.gov, January 16, 2019. https://www.energy.gov/ne/articles/advanced-reactor-technology-development-fact-sheet.
DOE-NE. 2020. "Infographic: Advanced Reactor Development" December 15, 2020. https://www.energy.gov/ne/articles/infographic-advanced-reactor-development.
DOE-NE. 2021a. "Office of Nuclear Energy: Strategic Vision." January 8, 2021. https://www.energy.gov/sites/prod/files/2021/01/f82/DOE-NE%20Strategic%20Vision%20-Web%20-%2001.08.2021.pdf
DOE-NE. 2021b. "What Is a Nuclear Microreactor?" February 26, 2021, https://www.energy.gov/ne/articles/what-nuclear-microreactor.
DOE-NE. 2021c. "Advanced Small Modular Reactors (SMRs)." https://www.energy.gov/ne/advanced-small-modular-reactors-smrs. Accessed December 28, 2021.
DOE-NE. 2021d. "Next-Gen Nuclear Plant and Jobs are Coming to Wyoming." November 16, 2021. https://www.energy.gov/ne/articles/next-gen-nuclear-plant-and-jobs-are-coming-wyoming.
DOE-NE. 2021e. "U.S. Department of Energy Seeks Input on Creation of HALEU Availability Program." Energy.gov, December 14, 2021. https://www.energy.gov/ne/articles/us-department-energy-seeks-input-creation-haleu-availability-program.
DOE-NE. 2021f. "Centrus Becomes First U.S. Licensed HALEU Production Facility." Energy.gov, June 23, 2021. https://www.energy.gov/ne/articles/centrus-becomes-first-us-licensed-haleu-production-facility.
DOE-NE. 2021g. "Request for Information (RFI) Regarding Planning for Establishment of a Program to Support the Availability of High-Assay Low-Enriched Uranium (HALEU) for Civilian Domestic Research, Development, Demonstration, and Commercial Use." Federal Register 86(237). December 14, 2021. https://www.govinfo.gov/content/pkg/FR-2021-12-14/pdf/2021-26984.pdf.
DOE-NE. 2021h. Federal Register. 2021. "Notice of Request for Information (RFI) on Using a Consent-Based Siting Process to Identify Federal Interim Storage Facilities," December 1, 2021. https://www.federalregister.gov/documents/2021/12/01/2021-25724/notice-of-request-for-information-rfi-on-using-a-consent-based-siting-process-to-identify-federal.
DOE-NE. 2022a. "U.S. Department of Energy Selects New Members to Nuclear Advisory Team, Revises Function and Focus." Energy.gov, February 9, 2022. https://www.energy.gov/ne/articles/us-department-energy-selects-new-members-nuclear-advisory-team-revises-function-and.
DOE-NE. 2022b. "Final Versatile Test Reactor Environmental Impact Statement: Volume 2, Appendices A-H." DOE/EIS-0542. Idaho Falls, ID: U.S. Department of Energy. https://www.energy.gov/sites/default/files/2022-05/final-eis-0542-versatile-test-reactor-vol-2-2022-05.pdf.
DOE-NE. n.d.-a. "Bilateral Cooperation." Energy.gov. https://www.energy.gov/ne/nuclear-reactor-technologies/international-nuclear-energy-policy-and-cooperation/bilateral. Accessed February 23, 2022.
DOE-NE. n.d.-b. "Nuclear Energy Advisory Committee." https://www.energy.gov/ne/nuclear-energy-advisory-committee. Accessed January 1, 2022.
DOE-NNSA (National Nuclear Security Administration). n.d.-a. "Office of Secure Transportation." Energy.gov. https://www.energy.gov/nnsa/office-secure-transportation.

DOE-NNSA. n.d.-b. "National Nuclear Security Administration." Energy.gov. https://www.energy.gov/nnsa/national-nuclear-security-administration.

DOE-RW (Office of Civilian Radioactive Waste Management). 2008. "Report to Congress on the Demonstration of the Interim Storage of Spent Nuclear Fuel from Decommissioned Nuclear Power Reactor Sites." DOE/RW-0596. Washington, DC: U.S. Department of Energy, December 2008. https://www.nrc.gov/docs/ML0834/ML083450160.pdf.

DOS (U.S. Department of State). 2010. Treaty on the Non-Proliferation of Nuclear Weapons. October 6. https://www.state.gov/wp-content/uploads/2019/03/228-Nuclear-Nonproliferation-Treaty-508.pdf. Accessed April 21, 2022.

DOS. n.d. "Nuclear Non-Proliferation Treaty." U.S. Department of State. https://www.state.gov/nuclear-nonproliferation-treaty.

Drain, F., J. L. Emin, R. Vinoche, and P. Baron. 2008. "COEX - Process: Cross-Breeding Between Innovation and Industrial Experience." Proceedings of Waste Management Symposium 2008. Tempe, AZ, United States. https://www.osti.gov/biblio/21319762.

Dungan, K., G. Butler, F. R. Livens, and L. M. Warren. 2017. "Uranium from Seawater—Infinite Resource or Improbable Aspiration?" Progress in Nuclear Energy 99(August 1):81-85. https://doi.org/10.1016/j.pnucene.2017.04.016.

Durst, P. C., I. Therios, R. Bean, A. Dougan, B. Boyer, R. Wallace, M. H Ehinger, D. N. Kovacic, and K. Tolk. 2007. "Advanced Safeguards Approaches for New Reprocessing Facilities." PNNL-16674. Richland, WA: Pacific Northwest National Laboratory, June 24, 2007. https://doi.org/10.2172/949149.

Durst, P. C., D. Beddingfield, B. Boyer, R. Bean, M. Collins, M. Ehinger, D. Hanks, D. L Moses, and L. Refalo. 2009. "Nuclear Safeguards Considerations for the Pebble Bed Modular Reactor (PBMR)." INL/EXT-09-16782. Idaho Falls, ID: Idaho National Laboratory, October 1, 2009. https://doi.org/10.2172/968683.

Ebbinghaus, B. B., C. G. Bathke, D. S. Dalton, and J. P. Murphy. 2013. "The Application of Material Attractiveness in a Graded Approach to Nuclear Materials Security." LLNL-CONF-640708. Salt Lake City, UT. https://www.osti.gov/biblio/1097225.

EIA (U.S. Energy Information Administration). 2020. "2019 Domestic Uranium Production Report," May 2020. https://www.eia.gov/uranium/production/annual/pdf/uprodshiptbl3.pdf.

EIA. 2021a. "Nuclear Explained: Where Our Uranium Comes From." U.S. Energy Information Administration, July 2, 2021. https://www.eia.gov/energyexplained/nuclear/where-our-uranium-comes-from.php.

EIA. 2021b. "Annual Energy Outlook 2021 with Projections to 2050." AEO 2021 Narrative. Washington, DC: U.S. Energy Information Administration, February 2021. https://www.eia.gov/outlooks/aeo/pdf/AEO_Narrative_2021.pdf.

EIA. 2021c. "Nuclear Explained: U.S. Nuclear Industry." U.S. Energy Information Administration, April 6, 2021. www.eia.gov/energyexplained/nuclear/us-nuclear-industry.php.

EIA. 2022. "Table 8.1 Nuclear Energy Overview," February 2022. https://www.eia.gov/totalenergy/data/monthly/pdf/sec8_3.pdf.

Energy Fuels. 2018. "Ur-Energy and Energy Fuels Announce U.S. Department of Commerce Has Initiated Investigation into Effects of Uranium Imports on U.S. National Security." Energy Fuels, July 18, 2018. https://www.energyfuels.com/2018-07-18-Ur-Energy-and-Energy-Fuels-Announce-U-S-Department-of-Commerce-Has-Initiated-Investigation-into-Effects-of-Uranium-Imports-on-U-S-National-Security.

ENS (Environment News Service). 2022. "UK Local Governments Demand Plutonium Safety Upgrades." ENS Newswire, February 12, 2022. https://ens-newswire.com/uk-local-governments-demand-plutonium-safety-upgrades.

EPRI (Electric Power Research Institute). 1991. "An Evaluation of the Concept of Transuranic Burning Using Liquid Metal Reactors." Electric Power Research Institute. http://inis.iaea.org/search/search.aspx?orig_q=RN:22056446.

EPRI. 2007. "An Economic Analysis of Select Fuel Cycles Using the Steady-State Analysis Model for Advanced Fuel Cycles Schemes (SMAFS)." Technical Update 1015387. Palo Alto, CA: Electric Power Research Institute.

EPRI. 2009a. "Program on Technology Innovation: Readiness of Existing and New U.S. Reactors for Mixed-Oxide (MOX) Fuel." Technical Update. Palo Alto, CA: Electric Power Research Institute, May 2009. https://www.epri.com/research/products/1018896.

EPRI. 2009b. Electric Power Research Institute. 2009. "Nuclear Fuel Cycle Cost Comparison Between Once-Through and Plutonium Single-Recycling in Pressurized Water Reactors." 1018575. Palo Alto, CA: Electric Power Research Institute. https://www.epri.com/research/products/1018575.

EPRI. 2010a. "Parametric Study of Front-End Nuclear Fuel Cycle Costs Using Reprocessed Uranium." No. 1020659. Technical Report. Palo Alto, CA: Electric Power Research Institute, January 2010.

EPRI. 2010b. "Advanced Nuclear Fuel Cycles—Main Challenges and Strategic Choices, Report No. 1020307, A. Machiels and A. Sowder. Palo Alto, CA: Electric Power Research Institute, September 2010. https://www.epri.com/research/products/1020307.

EPRI. 2010c. "Nuclear Fuel Cycle Cost Comparison Between Once-Through and Plutonium Multi-Recycling in Fast Reactors." 1020660. Palo Alto, CA: Electric Power Research Institute. https://www.epri.com/research/products/1020660.

EPRI. 2010d. "Nuclear Fuel Cycle Cost Comparison Between Once-Through and Fully Closed Cycles." 1021054. Palo Alto, CA: Electric Power Research Institute.

EPRI. 2019. "Uranium Oxycarbide (UCO) Tristructural Isotropic (TRISO) Coated Particle Fuel Performance." Technical Report EPRI-AR-1(NP). Palo Alto, CA: Electric Power Research Institute, May 2019. https://www.nrc.gov/docs/ML1915/ML19155A173.pdf.

EPRI. 2020. "Feasibility of Borehole Co-Location with Advanced Reactors for Onsite Management of Spent Nuclear Fuel." Technical Report 3002019751. Palo Alto, CA: Electric Power Research Institute. https://www.epri.com/research/products/000000003002019751.

EPRI. 2021a. "Evaluation of Chloride Fuel Salt Lifetime in a Fast-Spectrum, Liquid-Fuel Molten Salt Reactor." 3002021038. Technical Report. Palo Alto, CA: Electric Power Research Institute. https://www.epri.com/research/products/000000003002021038.

EPRI. 2021b. "Extended Storage Collaboration Program (ESCP) Winter 2021." EPRI. 2021. https://www.epri.com/events/8B781C3A-CD5F-48DC-96BA-9B570F6FD432.

EPRI. 2022. "Program 41.03.01: Used Fuel and High-Level Waste Management Program—Overview." EPRI. 2022. https://www.epri.com/research/programs/061149/overview.

Evans-Worrall, L. 2021. "Safeguards Technology Considerations and Research Needs for Thorium Fuel Cycles and Molten Salt Reactors." Presentation to the Committee on Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Reactors, October 15, virtual meeting. https://www.nationalacademies.org/event/10-14-2021/merits-and-viability-of-different-nuclear-fuel-cycles-and-technology-options-and-the-waste-aspects-of-advanced-nuclear-reactors-meeting-11-public-sessions-october-14-15-2021.

Ewing, R.C. 1995. "Radioactive Waste Forms: A Review and Comparison." In *Radioactive Waste Management Practices and Issues in Developing Countries: Proceedings of a Seminar Held in Beijing, China, 10-14 October 1994*. IAEA-TECDOC-851. Vienna, Austria: International Atomic Energy Agency. https://inis.iaea.org/collection/NCLCollectionStore/_Public/27/042/27042093.pdf.

Ewing, R. C. 2011. "Standards and Regulations for the Geologic Disposal of Spent Nuclear Fuel and High-Level Waste." Prepared for the Blue Ribbon Commission on America's Nuclear Future. Ann Arbor, MI: University of Michigan. https://fsi-live.s3.us-west-1.amazonaws.com/s3fs-public/ewing_brc_white_paper_final.pdf.

Ewing, R. C. 2015. "Long-Term Storage of Spent Nuclear Fuel." Nature Materials 14(March 2015):252-257. https://doi.org/10.1038/nmat4226.

Ewing, R. C., and L. M. Wang. 2002. "Phosphates as Nuclear Waste Forms." Reviews in Mineralogy and Geochemistry 48(1):673-699. https://doi.org/10.2138/rmg.2002.48.18.

Ewing, R. C., and S. Park. 2021. "The Concept of Geological Disposal of Highly Radioactive Nuclear Waste." Encyclopedia of Nuclear Energy. https://doi.org/10.1016/B978-0-12-819725-7.00156-2.

Fachinger, J., M. den Exter, B. Grambow, S. Holgersson, C. Landesman, M. Titov, and T. Podruhzina. 2006. "Behaviour of Spent HTR Fuel Elements in Aquatic Phases of Repository Host Rock Formations." HTR-2004 236(5) (March 1):543-554. https://doi.org/10.1016/j.nucengdes.2005.11.023.

Ferroni, P., F. Franceschini, and A. Levinsky. 2021. Updated information from "Westinghouse Lead Fast Reactor." Presentation to the Committee on Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Reactors, October 15, virtual meeting. https://www.nationalacademies.org/event/10-14-2021/merits-and-viability-of-different-nuclear-fuel-cycles-and-technology-options-and-the-waste-aspects-of-advanced-nuclear-reactors-meeting-11-public-sessions-october-14-15-2021.

Fisher, M. J., M. H. Holmes, and S. W. Chestnutt. 1996. "Fort St. Vrain Decommissioning Experience." Transactions of the American Nuclear Society 75(December 31, 1996). https://www.osti.gov/biblio/426590.

Flanagan, G., T. Fanning, and T. Sofu. 2015. "Sodium-Cooled Fast Reactor (SFR) Technology and Safety Overview." Presented at the Sodium-Cooled Fast Reactor Technology Seminar, Washington, DC, February 18, 2015. https://gain.inl.gov/SiteAssets/Fast%20Reactors/SFR-NRCTechnologyandSafetyOverview18Feb15.pdf.

Fletcher, D. 2020. "URENCO Next Generation Fuels: Conversion and Enrichment Options." Presented at the HALEU Webinar, April 28. https://gain.inl.gov/HALEU_Webinar_Presentations/11-Fletcher,URENCO-28Apr2020.pdf.

Flibe Energy. 2022. "LFTR Technology." Flibe Energy, 2022. https://flibe-energy.com/lftr.

Forsberg, C. W. 2007. "Thermal- and Fast-Spectrum Molten Salt Reactors for Actinide Burning and Fuel Production." In Proceedings of GLOBAL 2007 Conference on Advanced Nuclear Fuel Cycles and Systems. Boise, ID: American Nuclear Society, September 9-13, 2007.

Forsberg, C. W., and E. Greenspan. 2003. "Molten Salt Reactors (MSRs): Coupling Spent Fuel Processing and Actinide Burning." In Advances in Nuclear Fuel Management III (ANFM 2003). Hilton Head, SC: American Nuclear Society. https://technicalreports.ornl.gov/cppr/y2001/pres/118013.pdf.

Forsberg, C. and P. F. Peterson. 2015. Spent Nuclear Fuel and Graphite Management for Salt-Cooled Reactors: Storage, Safeguards, and Repository Disposal. Nuclear Technology 191(2):113-121. https://doi.org/10.13182/NT14-88.

Forsberg, C. W., C. M. Hopper, J. L. Richter, and H. C. Vantine. 1998. "Definition of Weapons-Usable Uranium-233." ORNL/TM-13517. Oak Ridge, TN: Oak Ridge National Laboratory. http://inis.iaea.org/search/search.aspx?orig_q=RN:29058723.

Forsberg, C. W., S. Lam, D. M. Carpenter, D. G. Whyte, R. Scarlat, C. Contescu, L. Wei, J. Stempien, and E. Blandford. 2017. "Tritium Control and Capture in Salt-Cooled Fission and Fusion Reactors: Status, Challenges, and Path Forward." Nuclear Technology 197(2):119-139. https://doi.org/10.13182/NT16-101.

Fort, J. A., D. J. Richmond, B. J. Jensen, and S. R. Suffield. 2019. "High-Burnup Demonstration: Thermal Modeling of TN-32B Vacuum Drying and ISFSI Transients." Richland, WA: Pacific Northwest National Laboratory, September 20, 2019. https://doi.org/10.2172/1568885.

Frank, S. M., and M. N. Patterson. 2014. "Electrorefiner Salt Disposition Technical Evaluation." TEV-2201. Idaho National Laboratory.

Frank, S., W. Ebert, B. Riley, H. Seo Park, C. H. Lee, M. K. Jeon, J. H. Yang, and H. C. Eun. 2015. "Waste Stream Treatment and Waste Form Fabrication for Pyroprocessing of Used Nuclear Fuel." INL/EXT-14-34014. Idaho Falls, ID: Idaho National Laboratory, July 2015. https://inldigitallibrary.inl.gov/sites/sti/sti/6597429.pdf.

Fredrickson, G. L., G. Cao, R. Gakhar, and T.-S. Yoo. 2018. "Molten Salt Reactor Salt Processing—Technology Status." INL/EXT-18-51033-Rev000. Idaho Falls, Idaho: Idaho National Laboratory, August 29, 2018. https://doi.org/10.2172/1484689.

Freeze, G., E. Bonano, P. Swift, E. Kalinina, E. Hardin, L. Price, S. Durbin, R. Rechard, and K. Gupta. 2021. "Integration of the Back End of the Nuclear Fuel Cycle." SAND2021-10444. Albuquerque, NM: Sandia National Laboratory, August 1, 2021. https://doi.org/10.2172/1817319.

Frieß, F., M. Kütt, and M. Englert. 2015. "Proliferation Issues Related to Fast SMRs." Annals of Nuclear Energy (Oxford) 2015:725-731.

FRWG (Fast Reactor Working Group). 2018. "Nuclear Metal Fuel: Characteristics, Design, Manufacturing, Testing, and Operating History," June 2018. https://www.nrc.gov/docs/ML1816/ML18165A249.pdf.

Fuks, L., I. Herdzik-Koniecko, K. Kiegiel, and G. Zakrzewska-Koltuniewicz. 2020. "Management of Radioactive Waste Containing Graphite: Overview of Methods." Energies 13(18). https://doi.org/10.3390/en13184638.

Fukuda, K., T. Ogawa, S. Kashimura, K. Ikawa, K. Iwamoto, K. Yamamoto, T. Itoh, and H. Matsushima. 1982. "Release Behavior of Fission Gas from Coated Fuel Particles under Irradiation." Journal of Nuclear Science and Technology 19(11):889-902. https://doi.org/10.1080/18811248.1982.9734235.

GA (General Atomics). 2019. "Status Report—EM2." Advanced Reactors Information System. International Atomic Energy Agency. September 30, 2019. https://aris.iaea.org/PDF/EM2(GeneralAtomics)_2020.pdf.

GAO (U.S. Government Accountability Office). 1977. "Issues Related to the Closing of the Nuclear Fuel Services, Incorporated, Reprocessing Plant at West Valley, New York." Washington, DC. https://www.gao.gov/assets/emd-77-27.pdf.

GAO. 2011. Commercial Nuclear Waste: Effects of a Termination of the Yucca Mountain Repository Program and Lessons Learned. (GAO-11-229). Washington, DC: U.S. Government Accountability Office. https://www.gao.gov/assets/gao-11-229.pdf. Accessed March 15, 2022.

GAO. 2013a. "Managing Critical Isotopes: Stewardship of Lithium-7 Is Needed to Ensure a Stable Supply." GAO-13-716. Washington, DC: U.S. Government Accountability Office. https://www.gao.gov/assets/gao-13-716.pdf.

GAO. 2013b. "Department of Energy: Concerns with Major Construction Projects at the Office of Environmental Management and NNSA." GAO-13-484T. Washington, DC: U.S. Government Accountability Office, March 20, 2013. https://www.gao.gov/assets/gao-13-484t.pdf.

GAO. 2017. Benefits and Costs Should Be Better Understood Before DOE Commits to a Separate Repository for Defense Waste. Report to congressional addressees. GAO-17-174. https://www.gao.gov/assets/gao-17-174.pdf.

GAO. 2018. "NNSA Should Clarify Long-Term Uranium Enrichment Mission Needs and Improve Technology Cost Estimates," GAO-18-126, Washington, DC: U.S. Government Accountability Office, February 2018, https://www.gao.gov/assets/gao-18-126.pdf.

GAO. 2019. "Surplus Plutonium Disposition: NNSA's Long-Term Plutonium Oxide Production Plans Are Uncertain." GAO-20-166. Washington, DC: U.S. Government Accountability Office, October 2019. https://www.gao.gov/assets/gao-20-166.pdf.

GAO. 2020. "Hanford Waste Treatment Plant: DOE Is Pursuing Pretreatment Alternatives, But Its Strategy Is Unclear While Costs Continue to Rise" (Accessible Version) (GAO-20-363). Washington, DC: U.S. Government Accountability Office. https://www.gao.gov/assets/710/707110.pdf. Accessed March 15, 2022.

GAO. 2021a. "Commercial Spent Nuclear Fuel: Congressional Action Needed to Break Impasse and Develop a Permanent Solution." Washington, DC: U.S. Government Accountability Office, September 2021. https://www.gao.gov/assets/gao-21-603.pdf.

GAO. 2021b. "Nuclear Waste Disposal: Actions Needed to Enable DOE Decision That Could Save Tens of Billions of Dollars." GAO-22-104365. Washington, DC: U.S. Government Accountability Office, December 2021. https://www.gao.gov/assets/gao-22-104365.pdf.

Garrett, A. G., S. L. Garrett, R. J. Marek, M. R. Mitchell, C. R. Orton, R. T. Otto, T. Sobolev, and D. C. Springfels. 2021. "Advanced Reactor Safeguards: Lessons from the IAEA Safeguards Domain." PNNL-31977. Richland, WA: Pacific Northwest National Laboratory, September 2021. https://gain.inl.gov/SiteAssets/AdvancedReactorSafeguards/KeyThrustArea5/PNNL-31977_ARS_IAEA_Interfaces.pdf.

Gay, A. 2021. "Back-End of the Nuclear Fuel Cycle: Existing Recycling Solution at Orano." Presentation to the Committee on Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Reactors, September 15, virtual meeting. https://www.nationalacademies.org/event/09-13-2021/merits-and-viability-of-different-nuclear-fuel-cycles-and-technology-options-and-the-waste-aspects-of-advanced-nuclear-reactors-meeting-9-september-13-15-2021.

Gerczak, T. J., R. L. Seibert, J. D. Hunn, C. A. Baldwin, F. C. Montgomery, and R. N. Morris. 2020. "Redistribution of Radionuclides in Irradiated AGR-1 UCO TRISO Fuel after 1800°C Safety Testing." Journal of Nuclear Materials 542(December 15):152453. https://doi.org/10.1016/j.jnucmat.2020.152453.

GIF (Generation IV International Forum). 2021a. "Sodium-Cooled Fast Reactor (SFR)." Generation IV Systems, 2021. https://www.gen-4.org/gif/jcms/c_9361/sfr.

GIF. 2021b. "Gas-Cooled Fast Reactor (GFR)." Generation IV Systems, 2021. https://www.gen-4.org/gif/jcms/c_42148/gas-cooled-fast-reactor-gfr.

GIF. 2021c. "Lead-Cooled Fast Reactor (LFR)." Generation IV Systems, 2021. https://www.gen-4.org/gif/jcms/c_42149/lead-cooled-fast-reactor-lfr.

GIF. 2021d. "Molten Salt Reactor." Generation IV Systems, 2021. https://www.gen-4.org/gif/jcms/c_9359/msr.

GIF. 2022a. "Very-High-Temperature Reactor." Generation IV International Forum. https://www.gen-4.org/gif/jcms/c_42153/very-high-temperature-reactor-vhtr.

GIF. 2022b. Annual Report 2021. https://www.gen-4.org/gif/jcms/c_203440/gif-2021-ar.

GIF and NERAC (U.S. Nuclear Energy Research Advisory Committee). 2002. "A Technology Roadmap for Generation IV Nuclear Energy Systems: Ten Nations Preparing Today for Tomorrow's Energy Needs." GIF-002-00. https://www.gen-4.org/gif/upload/docs/application/pdf/2013-09/genivroadmap2002.pdf.

GIF Experts' Group. 2010. "Use of Thorium in the Nuclear Fuel Cycle: How Should Thorium Be Considered in the GIF?" Generation IV International Forum, December 23, 2010. https://www.gen-4.org/gif/upload/docs/application/pdf/2013-10/gif_egthoriumpaperfinal.pdf.

Giraud, A. L. 1983. "Energy in France." Annual Review of Energy and the Environment 8:165-191. https://www.annualreviews.org/doi/pdf/10.1146/annurev.eg.08.110183.001121.

Gitau, E. T., and W. S. Charlton. 2012. "Use of a Microsphere Fingerprint for Identity Verification of Fuel Pebbles in a Pebble-Fueled HTGR." Journal of Nuclear Materials Management 40(2):19-24.

Glaser, A. 2006. "On the Proliferation Potential of Uranium Fuel for Research Reactors at Various Enrichment Levels." Science and Global Security 14(1):1-24. https://doi.org/10.1080/08929880600620542.

Goldner, A. and S. McKee. 2021. "Perspectives from Congress: Panel Discussion with Key Staff of the Appropriations Committees of the Senate and House of Representatives." Presentation to the Committee on Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Reactors, December 6, virtual meeting. https://www.nationalacademies.org/event/12-06-2021/merits-and-viability-of-different-nuclear-fuel-cycles-and-technology-options-and-the-waste-aspects-of-advanced-nuclear-reactors-meeting-12-december-6-7-2021.

Gougar, H. D. 2018. "Gas-Cooled Fast Reactor Research and Development Roadmap." United States: Idaho National Laboratory. May 3. https://doi.org/10.2172/1599068.

Grambow, B. 2008. "Mobile Fission and Activation Products in Nuclear Waste Disposal." Radiogeochemical Aspects of Nuclear Waste Disposal 102(3):180-186. https://doi.org/10.1016/j.jconhyd.2008.10.006.

Grambow, B. 2021. "Some Impacts of Advanced Fuel Cycle Options on Waste Management and Long Term Disposal Safety." Presentation to the Committee on Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Reactors, June 7, virtual meeting. https://www.nationalacademies.org/event/06-07-2021/merits-and-viability-of-different-nuclear-fuel-cycles-and-technology-options-and-the-waste-aspects-of-advanced-nuclear-reactors-meeting-6-part-2-june-7-2021.

Grambow, B., A. Loida, A. Martinez-Esparza, P. Diaz-Arocas, J. de Pablo, J-L. Paul, G. Marx, J.-P. Glatz, K. Lemmens, K. Ollila, and H. Christensen. 2000. "Source Term for Performance Assessment of Spent Fuel as a Waste Form." EUR19140 Report. Nuclear Science and Technology. European Commission. https://www.nrc.gov/docs/ML0334/ML033460436.pdf.

Grambow, B., A. Abdelouas, D. Greneche, M.H. Mouliney, J. Fachinger, A. Bukaemsky, S. Neumann, and M. Titov. 2006. "Study of Various Options for Final Disposal of HTR Coated Particles." In Proceedings HTR2006. Johannesburg, South Africa, 2006.

Grambow, B., A. Abdelouas, F. Guittonneau, J. Vandenborre, J. Fachinger, W. von Lensa, P. Bros, D. Roudil, J. Perko, J. Marivoet, A. Sneyers, D. Millington, and F. Cellier. 2008. "The Backend of the Fuel Cycle of HTR/VHTR Reactors." In Proceedings of the Fourth International Topical Meeting on High Temperature Reactor Technology 2:649-657. Washington, DC. https://doi.org/10.1115/HTR2008-58177.

Grambow, B., A. Abdelouas, F. Guittonneau, J. Vandenborre, W. von Lensa, P. Bros, and D. Roudil. 2010. "The Option of Geological Disposal of Spent (V)HTR Fuel." In Proceedings of HTR 2010, Paper 266. Prague, Czech Republic.

Grebennikova, T., A. N. Jones, and C. A. Sharrad. 2021. "Electrochemical Decontamination of Irradiated Nuclear Graphite from Corrosion and Fission Products Using Molten Salt." Energy and Environmental Science 14(10):5501-5512. https://doi.org/10.1039/D1EE00332A.

Grenthe, I., J. Fuger, R. J. Konings, R. J. Lemire, A. B. Muller, C. Nguyen-Trung, and H. Wanner. 1992. Chemical thermodynamics of uranium (vol. 1) Amsterdam: Elsevier.

Griffith, A. 2020. "Overview of DOE-NE's Fuel Cycles Work." Presentation to the Committee on Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Reactors, September 21, virtual meeting. https://www.nationalacademies.org/event/09-21-2020/merits-and-viability-of-different-nuclear-fuel-cycles-and-technology-options-and-the-waste-aspects-of-advanced-nuclear-reactors-september-21-23-2020-meeting.

Griffith, A. 2021. "Office of Nuclear Energy HALEU Availability Program FY2022 Proposed Activities." Presentation to the Committee on the Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Nuclear Reactors, September 28, virtual meeting. https://www.nationalacademies.org/event/09-28-2021/merits-and-viability-of-different-nuclear-fuel-cycles-and-technology-options-and-the-waste-aspects-of-advanced-nuclear-reactors-meeting-10-september-28-29-2021-public-sessions.

Guidez, J. 2013. Phénix. Le Retour d'expérience. EDP Sciences.

Guidez, J. 2017. "Phenix and Superphenix Feedback Experience." Presented at the Gen IV International Forum Webinar, 2017. https://www.gen-4.org/gif/upload/docs/application/pdf/2017-12/gifiv_webinar_phenix_superphenix_final_guidez.pdf.

Guittonneau, F., A. Abdelouas, and B. Grambow. 2010. "HTR Fuel Waste Management: TRISO Separation and Acid-Graphite Intercalation Compounds Preparation." Journal of Nuclear Materials 407(2):71-77. https://doi.org/10.1016/j.jnucmat.2010.09.026.

Haghighi, M., M. Ford, R. Szozda, and M. Jugan. 2002. "Waste Stream Generated and Waste Disposal Plans for Molten Salt Reactor Experiment at Oak Ridge National Laboratory." WM'02 Conference, February 24-28, 2002. Tucson, AZ. https://digital.library.unt.edu/ark:/67531/metadc782704/m1/1.

Hall, N., X. He, and Y.-M. Pan. 2019a. "Disposal Options and Potential Challenges to Waste Packages and Waste Forms in Disposal of Spent (Irradiated) Advanced Reactor Fuel Types." San Antonio, TX: Center for Nuclear Waste Regulatory Analyses, December 2019. https://www.nrc.gov/docs/ML2023/ML20237F397.pdf.

Hall, N., X. He, Y.-M. Pan, and P. LaPlante. 2019b. "Potential Challenges with Storage of Spent (Irradiated) Advanced Reactor Fuel Types." San Antonio, TX: Center for Nuclear Waste Regulatory Analysis, August 2019. https://www.nrc.gov/docs/ML2002/ML20022A217.pdf.

Hall, N., X. He, Y.-M. Pan, and P. LaPlante. 2020. "Transportation Experience and Potential Challenges with Transportation of Spent (Irradiated) Advanced Reactor Fuel Types." Prepared for U.S. Nuclear Regulatory Commission. San Antonio, TX: Center for Nuclear Waste Regulatory Analysis, January 2020.

Hanks, D. H. 2021. "Implementation of IAEA Safeguards Within the United States." Presentation to the Presentation to the National Academies of Sciences, Engineering, and Medicine's Committee on Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Reactors, September 14, 2021. https://www.nationalacademies.org/event/09-13-2021/merits-and-viability-of-different-nuclear-fuel-cycles-and-technology-options-and-the-waste-aspects-of-advanced-nuclear-reactors-meeting-9-september-13-15-2021.

Harp, J. M., S. L. Hayes, P. G. Medvedev, D. L. Porter, and L. Capriotti. 2017. "Testing Fast Reactor Fuels in a Thermal Reactor: A Comparison Report." Idaho National Laboratory. NTRD-FUEL-2017-00148. September 2017. https://www.osti.gov/servlets/purl/1458766.

Hawari, A. I., and J. Chen. 2005. "Computational Investigation of On-Line Interrogation of Pebble Bed Reactor Fuel." IEEE Transactions on Nuclear Science 52(5):1659-1664. October. https://doi.org/10.1109/TNS.2005.856760.

Hedin, A. 1997. "Spent Nuclear Fuel—How Dangerous Is It?" Stockholm, Sweden: Swedish Nuclear Fuel and Waste Management Co., March.

Hejzlar, P. 2021. "Terrapower: Fuel Cycle and Waste Management." Presentation to the National Academies of Sciences, Engineering, and Medicine's Committee on Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Reactors, February 22.

Higgins, M., N. Shoman, and B. Cipiti. 2021 "Limitations of Overall Measurement Error for Molten Salt Reactors." In Proceedings of the INMM and ESARDA Joint Virtual Annual Meeting, August 23-26 and August 30-September 1. https://resources.inmm.org/sites/default/files/2021-09/a190.pdf.

Hill, R. 2021. "Sodium-Cooled Fast Reactor Technologies." Presentation to the National Academies of Sciences, Engineering, and Medicine's Committee on Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Reactors, October 14, 2021. https://www.nationalacademies.org/event/10-14-2021/merits-and-viability-of-different-nuclear-fuel-cycles-and-technology-options-and-the-waste-aspects-of-advanced-nuclear-reactors-meeting-11-public-sessions-october-14-15-2021.

Hogue, K. K., P. Gibbs, M. P. Dion, and M. Poore. "Domestic Safeguards Material Control and Accountancy Considerations for Molten Salt Reactors." ORNL/SPR/150504. Oak Ridge, TN: Oak Ridge National Laboratory, February 2021. https://gain.inl.gov/SiteAssets/2021-April_SafeguardsAndSecurityWorkshop/Reading/Domestic%20Safeguards%20Material%20Control%20and%20Accountancy%20Considerations%20for%20Molten%20Salt%20Reactors%20-%20ORNL.pdf.

Holcomb, D. 2017. "Module 2: Overview of MSR Technology and Concepts." Presentation for U.S. Nuclear Regulatory Commission Staff, Washington, DC, November 7-8, 2017. https://www.nrc.gov/docs/ML1733/ML17331B114.pdf.

Holcomb, D. E. 2015. "Molten Salt Reactors Today: Status and Challenges." Presented at the Workshop on MSR Technologies—Commemorating the 50th Anniversary of the Startup of the MSRE, Oak Ridge National Laboratory, October 15, 2015. https://msrworkshop.ornl.gov/msr2015.

Holcomb, D. E., G. F. Flanagan, B. W. Patton, J. C. Gehin, R. L. Howard, and T. J. Harrison. 2011. "Fast Spectrum Molten Salt Reactor Options." ORNL/TM-2011/105. Oak Ridge, TN: Oak Ridge National Laboratory, July 2011. https://info.ornl.gov/sites/publications/Files/Pub29596.pdf.

Hollenbach, D. F., and C. M. Hopper. 1997. "Criticality Safety Considerations for MSRE Fuel Drain Tank Uranium Aggregation." CONF-970926-4. Topical Meeting on Criticality Safety Challenges in the Next Decade, Chelan, Washington, September 7-11, 1997. https://www.osti.gov/biblio/515562.

Holt, M. 2021a. "Civilian Nuclear Waste Disposal." Washington, DC: Congressional Research Service, September 17, 2021. https://sgp.fas.org/crs/misc/RL33461.pdf.

Holt, M. 2021b. "Nuclear Energy: Overview of Congressional Issues." Washington, DC: Congressional Research Service, October 20. https://crsreports.congress.gov/product/pdf/R/R42853.

Holt, M., and M. B. D. Nikitin. 2017. "Mixed-Oxide Fuel Fabrication Plant and Plutonium Disposition: Management and Policy Issues." Washington, DC: Congressional Research Service, December 14, 2017. https://fas.org/sgp/crs/nuke/R43125.pdf.

Holzman, J., and H. Northey. 2022. "Could Russia's Invasion of Ukraine Revive U.S. Uranium Mining?" E&E News: Greenwire. March 18, 2022. https://www.eenews.net/articles/could-russias-invasion-of-ukraine-revive-u-s-uranium-mining.

Hombourger, B., J. Křepel, and A. Pautz. 2019. "Breed-and-Burn Fuel Cycle in Molten Salt Reactors." EPJ Nuclear Sciences & Technologies 5. https://doi.org/10.1051/epjn/2019026.

Hoyt, N. C., C. A. Launiere, and E. A. Stricker. 2021. "In-Process Monitoring of Molten Salt Composition by Voltammetry and Automated Sampling-Based Techniques." Journal of Nuclear Materials Management 49(1):87-98.

Huff, K. D. 2019. "Chapter One—Economics of Advanced Reactors and Fuel Cycles." In Storage and Hybridization of Nuclear Energy, edited by H. Bindra and S. Revankar, pp. 1-20. Academic Press. https://doi.org/10.1016/B978-0-12-813975-2.00001-6.

Huff, K. D. 2021. "Perspectives from the Department of Energy's Office of Nuclear Energy." Presented at the Committee on Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Reactors, December 6.

Humrickhouse, P. W. 2011. "HTGR Dust Safety Issues and Needs for Research and Development." INL/EXT-11-21097. Idaho Falls, ID: Idaho National Laboratory. June. https://inldigitallibrary.inl.gov/sites/sti/sti/5026004.pdf.

Hyatt, N. C. 2020. "Safe Management of the UK Separated Plutonium Inventory: A Challenge of Materials Degradation." NPJ Materials Degradation 4:28. https://doi.org/10.1038/s41529-020-00132-7.

IAEA (International Atomic Energy Agency). 1970. "Treaty on the Non-Proliferation of Nuclear Weapons," INFCIRC/140. April 22, 1970. https://www.iaea.org/sites/default/files/publications/documents/infcircs/1970/infcirc140.pdf.

IAEA. 1975. "The Physical Protection of Nuclear Material," INFCIRC/225. Vienna, Austria: International Atomic Energy Agency. https://www.iaea.org/sites/default/files/publications/documents/infcircs/1975/infcirc225.pdf.

IAEA. 1980. "The Convention on the Physical Protection of Nuclear Material." INFCIRC/274/Rev. 1. Vienna, Austria: International Atomic Energy Agency. https://www.iaea.org/sites/default/files/infcirc274r1.pdf.

IAEA. 1981. "The Text of the Agreement of 18 November 1977 Between the United States of America and the Agency for the Application of Safeguards in the United States of America." INFCIRC/288. Information Circular. Vienna, Austria: International Atomic Energy Agency. https://www.iaea.org/sites/default/files/infcirc288.pdf.

IAEA. 1982. Nuclear Power, the Environment and Man. Vienna, Austria: International Atomic Energy Agency. https://www.iaea.org/publications/3478/nuclear-power-the-environment-and-man.

IAEA. 1985. IAEA Safeguards: Implementation at Nuclear Fuel Cycle Facilities. Safeguards Information Series 6. IAEA/SG/INF/6. Vienna, Austria: International Atomic Energy Agency. https://www.iaea.org/publications/1070/iaea-safeguards-implementation-at-nuclear-fuel-cycle-facilities.

IAEA. 1986. Summary Report on the Post-Accident Review Meeting on the Chernobyl Accident. INSAG Series 1. Vienna: International Atomic Energy Agency. https://www.iaea.org/publications/3598/summary-report-on-the-post-accident-review-meeting-on-the-chernobyl-accident.

IAEA. 1989. Decontamination and Decommissioning of Nuclear Facilities (Final Report of Three Research Co-Ordination Meetings Held Between 1984 and 1987). TECDOC Series 511. Vienna, Austria: International Atomic Energy Agency. https://www.iaea.org/publications/766/decontamination-and-decommissioning-of-nuclear-facilities-final-report-of-three-research-co-ordination-meetings-held-between-1984-and-1987.

IAEA. 1992. Radioactive Waste Management: An IAEA Source Book, Non-serial Publications. Vienna, Austria: International Atomic Energy Agency. https://www.iaea.org/publications/3760/radioactive-waste-management-an-iaea-source-book.

IAEA. 1995. Principles of Radioactive Waste Management Safety Fundamentals. Safety Series 111-F. Vienna, Austria: International Atomic Energy Agency. https://www.iaea.org/publications/5192/principles-of-radioactive-waste-management-safety-fundamentals.

IAEA. 1996. Significant Incidents in Nuclear Fuel Cycle Facilities. TECDOC Series 867. Vienna, Austria: International Atomic Energy Agency. https://inis.iaea.org/collection/NCLCollectionStore/_Public/27/060/27060437.pdf.

IAEA. 1997. "Spent Fuel Management: Current Status and Prospects." Nuclear Fuel Cycle and Materials Section. Vienna, Austria: International Atomic Energy Agency. https://www-pub.iaea.org/MTCD/Publications/PDF/te_1006_prn.pdf.

IAEA. 1999a. Report on the Preliminary Fact Finding Mission Following the Accident at the Nuclear Fuel Processing Facility in Tokaimura, Japan. Non-Serial Publications. Vienna, Austria: International Atomic Energy Agency. https://www-pub.iaea.org/MTCD/Publications/PDF/TOAC_web.pdf.

IAEA. 1999b. State of the Art Technology for Decontamination and Dismantling of Nuclear Facilities. Technical Reports Series 395. Vienna, Austria: International Atomic Energy Agency. https://www.iaea.org/publications/5783/state-of-the-art-technology-for-decontamination-and-dismantling-of-nuclear-facilities.

IAEA. 2002. "IAEA Safeguards Glossary—2001 Edition." International Nuclear Verification Series No. 3. Vienna, Austria: International Atomic Energy Agency. https://www.iaea.org/sites/default/files/iaea_safeguards_glossary.pdf.

IAEA. 2004a. "Management of Waste Containing Tritium and Carbon-14." Technical Report Series No. 421. Vienna, Austria. International Atomic Energy Agency. https://www-pub.iaea.org/MTCD/publications/PDF/TRS421_web.pdf.

IAEA. 2004b. "Operational and Decommissioning Experience with Fast Reactors." Proceedings of a technical meeting. Vienna, Austria: International Atomic Energy Agency. https://www-pub.iaea.org/MTCD/Publications/PDF/te_1405_web.pdf.

IAEA. 2005a. Thorium Fuel Cycle—Potential Benefits and Challenges. TECDOC Series 1450. Vienna: International Atomic Energy Agency. https://www.iaea.org/publications/7192/thorium-fuel-cycle-potential-benefits-and-challenges.

IAEA. 2005b. "Amendment to the Convention on the Physical Protection of Nuclear Material." INFCIRC/274/Rev.1/Mod. 1. Vienna, Austria: International Atomic Energy Agency. https://www.iaea.org/sites/default/files/publications/documents/infcircs/1979/infcirc274r1m1c.pdf.

IAEA. 2005c. "Status and Trends in Spent Fuel Reprocessing." IAEA-TECDOC-1467. Vienna, Austria: International Atomic Energy Agency. https://www-pub.iaea.org/MTCD/Publications/PDF/te_1467_web.pdf.

IAEA. 2006. Characterization, Treatment and Conditioning of Radioactive Graphite from Decommissioning of Nuclear Reactors. TECDOC Series 1521. Vienna, Austria: International Atomic Energy Agency. https://www.iaea.org/publications/7579/characterization-treatment-and-conditioning-of-radioactive-graphite-from-decommissioning-of-nuclear-reactors.

IAEA. 2007a. "Management of Reprocessed Uranium: Current Status and Future Prospects." Nuclear Fuel Cycle and Materials Section. Vienna, Austria: International Atomic Energy Agency. https://www-pub.iaea.org/MTCD/Publications/PDF/te_1529_web.pdf.

IAEA. 2007b. Radioactive Sodium Waste Treatment and Conditioning. TECDOC Series 1534. Vienna, Austria: International Atomic Energy Agency. https://www.iaea.org/publications/7642/radioactive-sodium-waste-treatment-and-conditioning.

IAEA. 2008. "Spent Fuel Reprocessing Options." IAEA-TECDOC-1587. Vienna, Austria: International Atomic Energy Agency. https://www-pub.iaea.org/MTCD/publications/PDF/te_1587_web.pdf.

IAEA. 2009. "Use of Reprocessed Uranium: Challenges and Options." Nuclear Energy Series No. NF-T-4.4. https://www-pub.iaea.org/MTCD/Publications/PDF/Pub1411_web.pdf.

IAEA. 2011a. "Safeguards Techniques and Equipment: 2011 Edition." International Nuclear Verification Series No. 1 (Rev. 2). Vienna, Austria: International Atomic Energy Agency. https://www-pub.iaea.org/MTCD/Publications/PDF/nvs1_web.pdf.

IAEA. 2011b. "Nuclear Security Recommendations on Physical Protection of Nuclear Material and Nuclear Facilities." INFCIRC/225/Revision 5. IAEA Nuclear Security Series No. 13. Vienna, Austria: International Atomic Energy Agency. https://www-pub.iaea.org/MTCD/Publications/PDF/Pub1481_web.pdf.

IAEA. 2011c. "Nuclear Security Recommendations on Radioactive Material and Associated Facilities." Nuclear Security Series No. 14, Vienna, Austria: International Atomic Energy Agency. https://www-pub.iaea.org/MTCD/Publications/PDF/Pub1487_web.pdf.

IAEA. 2013a. "International Safeguards in Nuclear Facility Design and Construction." IAEA Nuclear Energy Series, No. NF-T-2.8. Vienna, Austria: International Atomic Energy Agency. https://www-pub.iaea.org/MTCD/Publications/PDF/Pub1600_web.pdf.

IAEA. 2013b. Legal and Institutional Issues of Transportable Nuclear Power Plants: A Preliminary Study. Nuclear Energy Series, NG-T-3.5. Vienna: International Atomic Energy Agency. https://www.iaea.org/publications/10516/legal-and-institutional-issues-of-transportable-nuclear-power-plants-a-preliminary-study.

IAEA. 2013c. Status of Fast Reactor Research and Technology Development. TECDOC Series 1691. Vienna: International Atomic Energy Agency. https://www-pub.iaea.org/MTCD/Publications/PDF/te_1691_web.pdf.

IAEA. 2014. International Safeguards in the Design of Nuclear Reactors. Nuclear Energy Series, NP-T-2.9. Vienna, Austria: International Atomic Energy Agency. https://www.iaea.org/publications/10710/international-safeguards-in-the-design-of-nuclear-reactors.

IAEA. 2015a. Status of Accelerator Driven Systems Research and Technology Development. TECDOC Series 1766. Vienna, Austria: International Atomic Energy Agency. https://www.iaea.org/publications/10870/status-of-accelerator-driven-systems-research-and-technology-development.

IAEA. 2015b. Security of Nuclear Material in Transport: Implementation Guide. IAEA Nuclear Security Series No. 26-G. https://www.iaea.org/publications/10792/security-of-nuclear-material-in-transpor. Accessed April 21, 2022.

IAEA. 2015c. "Security of Nuclear and Other Radioactive Material in Transport." Version 2. Document Preparation Profile, NST053. Vienna, Austria: International Atomic Energy Agency. https://www.iaea.org/sites/default/files/dpp_nst053_0.pdf.

IAEA. 2016a. "Functional Specifications for Nuclear Security Detection Equipment and Systems." Document Preparation Profile NST059. Vienna, Austria: International Atomic Energy Agency, Version 1, April 1, 2016. https://www.iaea.org/sites/default/files/dpp_nst059.pdf.

IAEA. 2016b. "Development, Use and Maintenance of Threat Assessment and Design Basis Threat." Document Preparation Profile, NST058. Vienna, Austria: International Atomic Energy Agency, Version 0, March 18, 2016. https://www.iaea.org/sites/default/files/dpp_nst058.pdf.

IAEA. 2017a. "International Safeguards in the Design of Fuel Fabrication Plants." IAEA Nuclear Energy Series, No. NF-T-4.7. Vienna, Austria: International Atomic Energy Agency. https://www-pub.iaea.org/MTCD/Publications/PDF/Pub1699_web.pdf.

IAEA. 2017b. "Safety of Nuclear Fuel Cycle Facilities." No. SSR-4 Pub. 1791. IAEA Safety Standards for Protecting People and the Environment. Vienna, Austria: International Atomic Energy Agency. https://www-pub.iaea.org/MTCD/Publications/PDF/PUB1791_web.pdf.

IAEA. 2018a. Decommissioning of Nuclear Power Plants, Research Reactors and Other Nuclear Fuel Cycle Facilities. Specific Safety Guides, SSG-47. Vienna, Austria: International Atomic Energy Agency. https://www.iaea.org/publications/12210/decommissioning-of-nuclear-power-plants-research-reactors-and-other-nuclear-fuel-cycle-facilities.

IAEA. 2018b. Physical Protection of Nuclear Material and Nuclear Facilities (Implementation of INFCIRC/225/Revision 5). Implementing Guides 27-G. Vienna, Austria: International Atomic Energy Agency. https://www.iaea.org/publications/11092/physical-protection-of-nuclear-material-and-nuclear-facilities-implementation-of-infcirc/225/revision-5.

IAEA. 2019a. "International Safeguards in the Design of Enrichment Plants." IAEA Nuclear Energy Series, No. NF-T-4.10. Vienna, Austria: International Atomic Energy Agency. https://www-pub.iaea.org/MTCD/Publications/PDF/PUB1865_web.pdf.

IAEA. 2019b. "International Safeguards in the Design of Reprocessing Plants." IAEA Nuclear Energy Series, No. NF-T-3.2. Vienna, Austria: International Atomic Energy Agency. https://www-pub.iaea.org/MTCD/publications/PDF/PUB1866_web.pdf.

IAEA. 2019c. Security of Radioactive Material in Use and Storage and of Associated Facilities. Implementing Guides 11-G (Rev.1). Vienna, Austria: International Atomic Energy Agency. https://www.iaea.org/publications/12360/security-of-radioactive-material-in-use-and-storage-and-of-associated-facilities.

IAEA. 2019d. "Identification and Categorization of Sabotage Targets, and Identification of Vital Areas at Nuclear Facilities." Document Preparation Profile, NST063; Version 1.2, November 1. Vienna, Austria: International Atomic Energy Agency. https://www.iaea.org/sites/default/files/dpp_nst063_1.pdf.

IAEA. 2019e. "Waste from Innovative Types of Reactors and Fuel Cycles: A Preliminary Study." IAEA Nuclear Energy Series. Vienna, Austria: International Atomic Energy Agency. https://www-pub.iaea.org/MTCD/Publications/PDF/PUB1822_web.pdf.

IAEA. 2020a. Security of Radioactive Material in Transport. Implementing Guides 9-G (Rev. 1). Vienna, Austria: International Atomic Energy Agency. https://www.iaea.org/publications/13400/security-of-radioactive-material-in-transport.

IAEA. 2020b. "The Safeguards Implementation Report for 2019." GOV/2020/9. Safeguards Implementation Reports. Vienna, Austria: International Atomic Energy Agency. https://armscontrollaw.files.wordpress.com/2020/05/sir-2019.pdf.

IAEA. 2020c. Preventive and Protective Measures Against Insider Threats. Implementing Guides 8-G (Rev. 1). Vienna, Austria: International Atomic Energy Agency. https://www.iaea.org/publications/12354/preventive-and-protective-measures-against-insider-threats.

IAEA. 2020d. Light Water Reactor Fuel Enrichment Beyond the Five Per Cent Limit: Perspectives and Challenges. TECDOC Series 1918. Vienna, Austria: International Atomic Energy Agency. https://www.iaea.org/publications/14697/light water-reactor-fuel-enrichment-beyond-the-five-per-cent-limit-perspectives-and-challenges.

IAEA. 2021a. National Nuclear Security Threat Assessment, Design Basis Threats and Representative Threat Statements. Implementing Guides 10-G (Rev. 1). Vienna, Austria: International Atomic Energy Agency. https://www.iaea.org/publications/13618/national-nuclear-security-threat-assessment-design-basis-threats-and-representative-threat-statements.

IAEA. 2021b. Enhancing Nuclear Security Culture in Organizations Associated with Nuclear and Other Radioactive Material. Technical Guidance 38-T. Vienna, Austria: International Atomic Energy Agency. https://www.iaea.org/publications/13405/enhancing-nuclear-security-culture-in-organizations-associated-with-nuclear-and-other-radioactive-material.

IAEA. 2021c. "Communication Received from Certain Member States Concerning Their Policies Regarding the Management of Plutonium." INFCIRC/549. Vienna, Austria: International Atomic IAEA. Energy, Electricity and Nuclear Power Estimates for the Period up to 2050. Reference Data Series 1. Vienna, Austria: International Atomic Energy Agency. https://www.iaea.org/publications/15028/energy-electricity-and-nuclear-power-estimates-for-the-period-up-to-2050.

IAEA. 2021d. Status and Trends in Pyroprocessing of Spent Nuclear Fuels. TECDOC Series 1967. Vienna, Austria: International Atomic Energy Agency. https://www.iaea.org/publications/11062/status-and-trends-in-pyroprocessing-of-spent-nuclear-fuels.

IAEA. 2021e. Handbook on the Design of Physical Protection Systems for Nuclear Material and Nuclear Facility; Technical Guidance. IAEA Nuclear Security Series No. 40-T.

IAEA. 2021f. "Communication Received from France Concerning Its Policies Regarding the Management of Plutonium." INFCIRC/549/Add.5/25. Statements on the Management of Plutonium and of High Enriched Uranium. Vienna, Austria: International Atomic Energy Agency. https://www.iaea.org/sites/default/files/publications/documents/infcircs/1998/infcirc549a5-25.pdf.

IAEA. 2022. "Live Chart of Nuclides." IAEA Nuclear Data Section. https://www-nds.iaea.org/relnsd/vcharthtml/VChartHTML.html.

IAEA. n.d.-a. "The IAEA and the Non-Proliferation Treaty." IAEA. https://www.iaea.org/topics/non-proliferation-treaty.

IAEA. n.d.-b. "Nuclear Security Series." IAEA. https://www.iaea.org/resources/nuclear-security-series.

IAEA. n.d.-c. "Safeguards Agreements." https://www.iaea.org/topics/safeguards-agreements. Accessed April 20, 2022.

IDEQ (Idaho Department of Environmental Quality). 1995. "1995 Settlement Agreement," October 16. https://www2.deq.idaho.gov/admin/LEIA/api/document/download/14673.

Ignatiev, V., O. Feynberg, I. Gnidoi, A. Merzlyakov, A. Surenkov, V. Uglov, A. Zagnitko, V. Subbotin, I. Sannikov, A. Toropov, V. Afonichkin, A. Bovet, V. Khokhlov, V. Shishkin, M. Kormilitsyn, A. Lizin, and A. Osipenko. 2014. Molten salt actinide recycler and transforming system without and with Th–U support: Fuel cycle flexibility and key material properties. Annals of Nuclear Energy 64:408-420. https://doi.org/10.1016/j.anucene.2013.09.004.

IN2P3 (Institut National de Physique Nucléaire et Physique des Particules). n.d.-a. "Spent Fuel Composition." https://www.radioactivity.eu.com/site/pages/Spent_Fuel_Composition.htm.

IN2P3. n.d.-b. "Long-Lived Fission Products." https://www.radioactivity.eu.com/site/pages/Long_Lived_Fission_Products.htm.

INL (Idaho National Laboratory). 2021a. "INL's Advanced Test Reactor." Factsheet. Updated 2021. https://factsheets.inl.gov/FactSheets/Advanced%20Test%20Reactor%20--%20Safety.pdf.

INL. 2021b. "Advanced Fuels Campaign Execution Plan." Idaho Falls, ID: Idaho National Laboratory, July 29, 2021. https://www.osti.gov/biblio/1813552.

IPFM (International Panel on Fissile Materials). 2014. "Sweden Wants to Transfer Ownership of 834 kg of Separated Plutonium to the United Kingdom." IPFM Blog. March 18. https://fissilematerials.org/blog/2014/03/sweden_wants_to_transfer_.html.

IPFM. 2020. "UK Sellafield Magnox Reprocessing Plant to Close in 2021, One Year Later Than Planned." IPFM Blog. August 12, 2020. http://fissilematerials.org/blog/2020/08/uk_sellafield_magnox_repr.html.

IPFM. 2021. "Countries: India." International Panel on Fissile Materials, August 31, 2021. https://fissilematerials.org/countries/india.html.

JAEA (Japan Atomic Energy Agency). n.d. "Reprocessing Technology Development." Nuclear Fuel Cycle Engineering Laboratories. https://www.jaea.go.jp/english/04/tokai-cycle/02.htm.

JNFL (Japan Nuclear Fuel Limited). 2020 "MOX Fuel Fabrication." December 16. https://www.jnfl.co.jp/en/business/mox.

JNFL. 2022. "Operation Plans for Rokkasho Reprocessing Plant and MOX Fuel Fabrication Plant." Japan Nuclear Fuel Limited, February 10, 2022. https://www.jnfl.co.jp/en/release/press/2021/detail/20220210-1.html.

Johnson, S. 2009. "The Safeguards at Reprocessing Plants Under a Fissile Material (Cutoff) Treaty." Research Report No. 6. Princeton, NJ: International Panel on Fissile Materials, February 2009. https://fissilematerials.org/library/rr06.pdf.

Johnstone, C. 2022. "French Regulator Quizzes EDF, Orano on Fuel Storage Capacity Limits," Nucleonics Week, March 16.

Joly, P., and E. Boo. 2015. "Roadmap: Actinide Separation Processes 2015." Safety of Actinide Separation Processes. European Commission.

Jorgensen, L. 2021. "ThorCon." Presented at the Committee on Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Reactors, January 11, 2021.

Kailas, S., M. Hemalatha, and A. Saxena. 2015. "Nuclear Transmutation Strategies for Management of Long-Lived Fission Products." Pramana 85(3):517-523. September 1. https://doi.org/10.1007/s12043-015-1063-z.

Kairos. 2021. "Hermes Construction Permit Application Accepted for Review by Nuclear Regulatory Commission." November 30, 2021. https://kairospower.com/external_updates/hermes-construction-permit-application-accepted-for-review-by-nuclear-regulatory-commission.

Kamide, H. 2021. "Overview of the Generation IV International Forum." Presentation to the National Academies' Committee on Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Reactors, September 28, 2021.

Katoh, Y., L. L. Snead, I. Szlufarska, and W. J. Weber. 2012. "Radiation Effects in SiC for Nuclear Structural Applications." Material Challenges for Advanced Nuclear Power Systems 16(3):143-152. https://doi.org/10.1016/j.cossms.2012.03.005.

Kilochytska, T. 2021. "IAEA Designates New Collaborating Centre in France for Graphite Reactor Decommissioning." International Atomic Energy Agency. September 30. https://www.iaea.org/newscenter/news/iaea-designates-new-collaborating-centre-in-france-for-graphite-reactor-decommissioning.

Kim, Y. I., S. J. Kim, Y. G. Kim, and Y. J. Kim. 1999. "Conceptual Design of KALIMER Uranium Metallic Fueled Core." KAERI/TR-1279/99. Taejon, Republic of Korea: Korea Atomic Energy Research Institute. http://inis.iaea.org/search/search.aspx?orig_q=RN:46037652.

Kovacic, D. N., P. W. Gibbs, and L. M. Scott. 2020. "Model MC&A for Pebble Bed Reactors (Technical Direction No. 5 Task 2.6 Letter Report)." ORNL/SPR-2019/1329. Oak Ridge, TN: Oak Ridge National Laboratory. March 1. https://doi.org/10.2172/1606926.

Kovacic, D., P. Gibbs, L. Worrall, R. Hunneke, J. Harp, and J. Hu. 2021. "Advanced Reactor Safeguards: Nuclear Material Control and Accounting for Pebble Bed Reactors." ORNL/SPR-2020-1849. Oak Ridge, TN: Oak Ridge National Laboratory. January. https://gain.inl.gov/SiteAssets/2021-April_SafeguardsAndSecurityWorkshop/Reading/Advanced%20Reactor%20Safeguards%20-%20Nuclear%20Material%20Control%20and%20Accounting%20for%20Pebble%20Bed%20Reactors.pdf.

Kozak, M., T. A. Feeney, C. D. Leigh, and H. W. Stockman. 1992. Performance assessment of the proposed disposal of depleted uranium as Class A Low-Level Waste, Sandia National Laboratories, Albuquerque, NM. https: //www.nrc.gov/docs/ML1018/ML101890179.pdf.

Krahn, S. L., A. G. Croff, B. L. Smith, J. H. Clarke, A. G. Sowder, and A. J. Machiels. 2014. "Evaluating the Collective Radiation Dose to Workers from the U.S. Once-Through Nuclear Fuel Cycle." Nuclear Technology 185(2):192-207. https://doi.org/10.13182/NT13-64.

Krall, L., T. McCartin, and A. Macfarlane. 2020. "Siting Deep Boreholes for Disposal of Radioactive Waste: Consequences for Tight Coupling between Natural and Engineered Systems." Environmental Science & Technology 54(2):629-646. https://doi.org/10.1021/acs.est.9b03440.

Kramer, D. 2021. "Idaho project tests the limits of DOE aid to advanced reactors." Physics Today. November. https://physicstoday.scitation.org/doi/10.1063/PT.3.4878.

Kramer, D. 2022. "DOE Prepares to Put a Nearly Completed Uranium Contract up for Bid." Physics Today, February. https://doi.org/10.1063/PT.6.2.20220223a.

Krass, A. S., P. Boskma, B. Elzen, and W. A. Smit. 1983. Uranium Enrichment and Nuclear Weapon Proliferation. Milton Park, UK: Taylor and Francis.

Kumar, K.V. Suresh, A. Babu, B. Anandapadmanaban, and G. Srinivasan. 2011. "Twenty Five Years of Operating Experience with the Fast Breeder Test Reactor." Asian Nuclear Prospects 2010 7(January 1):323-332. https://doi.org/10.1016/j.egypro.2011.06.042.

LaHaye, N. L., and D. E. Burkes. 2019. "Metal Fuel Fabrication Safety and Hazards." Richland, WA: Pacific Northwest National Laboratory. June. https://www.nrc.gov/docs/ML1919/ML19193A057.pdf.

Landais, P. 2021. "France's Strategy on the Back-End of the Fuel Cycle and the Management and Disposal of Radioactive Waste: Is Final Disposal Still the Achilles' Heel of the Nuclear Industry?" Presented at the Committee on Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Reactors, September 29. https://www.nationalacademies.org/event/09-28-2021/merits-and-viability-of-different-nuclear-fuel-cycles-and-technology-options-and-the-waste-aspects-of-advanced-nuclear-reactors-meeting-10-september-28-29-2021-public-sessions.

Larsen, R. K. 2021. "Aspects of Material Accounting and Control for Advanced Reactors." Presentation to the National Academies' Committee on Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Reactors, May 17, 2021. https://www.nationalacademies.org/event/05-17-2021/merits-and-viability-of-different-nuclear-fuel-cycles-and-technology-options-and-the-waste-aspects-of-advanced-nuclear-reactors-may-17-19-2021-meeting.

Larson, L. N. 2019. "The Front End of the Nuclear Fuel Cycle: Current Issues." Washington, DC: Congressional Research Service. July 29. https://fas.org/sgp/crs/nuke/R45753.pdf.

Larson, L. N. 2020. "Uranium Reserve Program Proposal: Policy Implications." Washington, DC: Congressional Research Service. April 16. https://sgp.fas.org/crs/nuke/IF11505.pdf.

Latkowski, J. 2021. "TerraPower's Molten Chloride Fast Reactor (MCFR)." Presentation to the National Academies' Committee on Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Reactors, February 22, 2021. https://www.nationalacademies.org/event/02-22-2021/merits-and-viability-of-different-nuclear-fuel-cycles-and-technology-options-and-the-waste-aspects-of-advanced-nuclear-reactors-february-22-and-23-2021-meeting.

LeadCold. 2021. "Breakthrough Innovation." https://www.leadcold.com/technology.html. Accessed December 28, 2021.

LeBlanc, D. 2010. "Denatured Molten Salt Reactors (DMSR): An Idea Whose Time Has Finally Come?" Atoms for Power, Health and the Environment 31st annual conference of the Canadian Nuclear Society and 34th annual conference of the Canadian Nuclear Society and Canadian Nuclear Association. Canada: Canadian Nuclear Society. https://twugbcn.files.wordpress.com/2011/03/1-d_leblanc_ovrvw_denmsr.pdf.

LeBlanc, D. 2021. "Leading the Way to a Bright Energy Future: Overview of the IMSR." Presented at the National Academies' Committee on Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Reactors, online, January 11. https://www.nationalacademies.org/event/01-11-2021/merits-and-viability-of-different-nuclear-fuel-cycles-and-technology-options-and-the-waste-aspects-of-advanced-nuclear-january-11-2021-and-january-13-2021-meeting.

Lee, J. C. 2020. Nuclear Reactor: Physics and Engineering. Hoboken, NJ: John Wiley and Sons.

Lee, W. E., M. Gilbert, S. T. Murphy, and R. W. Grimes. 2013. "Opportunities for Advanced Ceramics and Composites in the Nuclear Sector." Journal of the American Ceramic Society 96(7):2005-2030. https://doi.org/10.1111/jace.12406.

Les Échos. 2016. "Nucléaire: les arrêts de réacteur vont couter 1 milliard d'euros à EDF en 2016." www.lesechos.fr/2016/11/nucleaire-les-arrets-de-reacteur-vont-couter-1-milliard-deurosa-edf-en-2016-225670.

Liu, D., S. Knol, J. Ell, H. Barnard, M. Davies, J. A. Vreeling, and R. O. Ritchie. 2020. "X-Ray Tomography Study on the Crushing Strength and Irradiation Behaviour of Dedicated Tristructural Isotropic Nuclear Fuel Particles at 1000°C." Materials and Design 187(February 1):108382. https://doi.org/10.1016/j.matdes.2019.108382.

Lommers, L. 2021. "Framatome Steam Cycle High-Temperature Gas-Cooled Reactors." Presentation to the National Academies' Committee on Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Reactors, February 22, 2021. https://www.nationalacademies.org/event/02-22-2021/merits-and-viability-of-different-nuclear-fuel-cycles-and-technology-options-and-the-waste-aspects-of-advanced-nuclear-reactors-february-22-and-23-2021-meeting.

Long, J. T. 1967. Engineering for Nuclear Fuel Reprocessing. New York: Gordon and Breach Science Publishers.

Machiels, A., S. Massara, and C. Garzenne. 2009. "Dynamic Analysis of Deployment Scenario of Faster Burner Reactors in the U.S. Nuclear Fleet." GLOBAL 2009, Paris, France.

Magill, J., V. Berthou, D. Haas, J. Galy, R. Schenkel, H.-W. Wiese, G. Heusener, J. Tommasi, G. Youinou. 2003. Impact Limits of Partitioning and Transmutation Scenarios on Radiotoxicity of Actinides in Radioactive Waste. Nuclear Energy 42:263-277. https://www.researchgate.net/profile/J-Tommasi/publication/252069412_Impact_Limits_of_Partitioning_and_Transmutation_Scenarios_on_Radiotoxicity_of_Actinides_in_Radioactive_Waste/links/0f31752e0c63d46953000000/Impact-Limits-of-Partitioning-and-Transmutation-Scenarios-on-Radiotoxicity-of-Actinides-in-Radioactive-Waste.pdf.

Mahanty, B., A. S. Kanekar, S. A. Ansari, A. Bhattacharyya, and P. K. Mohapatra. 2019. "Separation of Neptunium from Actinides by Monoamides: A Solvent Extraction Study." Radiochimica Acta 107(5):369-376. https://doi.org/10.1515/ract-2018-3074.

Majumdar, D., S. N. Jahshan, C. M. Allison, P. Kuan, and T. R. Thomas. 1992. "Recycling of Nuclear Spent Fuel with AIROX Processing." DOE/ID-10423. Idaho Falls, ID: U.S. Department of Energy. December 1. https://doi.org/10.2172/10146308.

Malherbe, J. B. 2013. "Diffusion of Fission Products and Radiation Damage in SiC." Journal of Physics D: Applied Physics 46(47):473001. https://doi.org/10.1088/0022-3727/46/47/473001.

Mallapaty, S. 2021. "China Prepares to Test Thorium-Fuelled Nuclear Reactor." Nature, September 10, 2021. https://www.nature.com/articles/d41586-021-02459-w.

Martin, G., M. Guyot, F. Laugier, G. Senentz, G. Krivtchik, B. Carlier, D. Lecarpentier, F. Descamps, C. Chabert, and R. Eschbach. 2018. "French Scenarios Toward Fast Plutonium Multi-Recycling in PWR." ICAPP, Charlotte, United States. ffcea-01908271. https://hal-cea.archives-ouvertes.fr/cea-01908271/document. Accessed April 20, 2022.

McFarlane, J. 2021. "Hazards Associated with Molten Salt Reactor Systems." Presentation to the National Academies' Committee on Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Reactors. October 15. https://www.nationalacademies.org/event/10-14-2021/merits-and-viability-of-different-nuclear-fuel-cycles-and-technology-options-and-the-waste-aspects-of-advanced-nuclear-reactors-meeting-11-public--sessions-october-14-15-2021.

McFarlane, J., P. A. Taylor, D. E. Holcomb, and W. Poore III. 2019. "Review of Hazards Associated with Molten Salt Reactor Fuel Processing Operations." United States. https://doi.org/10.2172/1543201.

McFarlane, J., B. J. Riley, D. E. Holcomb, A. Lines, H. B. Andrews, S. A. Bryan, A.S. Chapel, N. D. B. Ezell, H. M. Felmy, M. S. Greenwood, P. W. Humrickhouse, and K. G. Myhre. 2020. "Molten Salt Reactor Engineering Study for Off-Gas Management: Nuclear Technology Research and Development." ORNL/TN-2020/1602, PNNL-30159. August 31. https://info.ornl.gov/sites/publications/Files/Pub142799.pdf. Accessed April 20, 2022.

McMillan, B. 2019. "Molten Salt Reactor Experiment Project Initiatives." Oak Ridge Office of Environmental Management. https://www.energy.gov/sites/prod/files/2019/11/f68/ORSSAB%20MSRE%20Presentation%20-%20Nov%202019.pdf. Accessed April 21, 2022.

METI (Ministry of Economy, Trade, and Industry). 2016. "Spent Nuclear Fuel Reprocessing Implementation Act." METI. October 20, 2016. https://www.meti.go.jp/english/policy/energy_environment/law/index.html.

Metlay, D. S. 2016. "Selecting a Site for a Radioactive Waste Repository: A Historical Analysis." Elements 12:269-274. https://doi.org/DOI: 10.2113/gselements.12.4.269.

Metlay, D. S. 2021. "Social Acceptability of Geologic Disposal." In Encyclopedia of Nuclear Energy, edited by E. Greenspan, pp. 684-697. Oxford, UK: Elsevier. https://doi.org/10.1016/B978-0-12-819725-7.00157-4.

Miguirditchian, M., and R. Taylor. 2021. "The Adaptation of Recycling Processes to Pu-Multi Recycling." In Encyclopedia of Nuclear Energy, edited by E. Greenspan, pp. 523-533. Oxford, UK: Elsevier. https://doi.org/10.1016/B978-0-12-819725-7.00228-2.

Miguirditchian, M., S. Grandjean, C. Sorel, V. Vanel, T. Delahaye, S. Pillon, L. Ramond, et al. 2017. "Advanced Concepts for Uranium and Plutonium Multi-Recycling." In GLOBAL 2017 International Nuclear Fuel Cycle Conference. Seoul, South Korea. https://hal.archives-ouvertes.fr/hal-02418116.

Miles, B. 2016. "Naval Spent Fuel Transportation." Nuclear Waste Technical Review Board Meeting presented at the Summer 2016 Board Meeting, Washington, DC, August 24. https://www.nwtrb.gov/docs/default-source/meetings/2016/august/miles.pdf?sfvrsn=12.

MIT (Massachusetts Institute of Technology). 2003. "The Future of Nuclear Power: An Interdisciplinary MIT Study." Cambridge, MA: Massachusetts Institute of Technology. https://web.mit.edu/nuclearpower/pdf/nuclearpower-full.pdf.

MIT. 2009. "Update of the MIT 2003 Future of Nuclear Power: An Interdisciplinary MIT Study." Cambridge, MA: Massachusetts Institute of Technology. https://web.mit.edu/nuclearpower/pdf/nuclearpower-update2009.pdf.

MIT. 2011. "The Future of the Nuclear Fuel Cycle." Cambridge, MA: Massachusetts Institute of Technology, 2011. https://energy.mit.edu/wp-content/uploads/2011/04/MITEI-The-Future-of-the-Nuclear-Fuel-Cycle.pdf.

MIT. 2018. "The Future of Nuclear Energy in a Carbon-Constrained World: An Interdisciplinary MIT Study." Cambridge, MA: Massachusetts Institute of Technology. http://energy.mit.edu/wp-content/uploads/2018/09/The-Future-of-Nuclear-Energy-in-a-Carbon-Constrained-World.pdf. MIT Energy Initiative

Moltex Energy. 2021. "Moltex Completes Phase 1 of CNSC Vendor Design Review." Moltex Energy, May 25, 2021. https://www.moltexenergy.com/moltex-completes-phase-1-of-cnsc-vendor-design-review.

Moltex Energy. 2022. "FAQ." Moltex Energy. https://www.moltexenergy.com/faq.

Moore, E. N., and T. F. Severynse. 2020. "Conceptual Assessment of VTR Add-on Processing Capability." Aiken, SC: Savannah River National Laboratory. July 6. https://web.archive.org/web/20201105184215id_/https://sti.srs.gov/fulltext/SRNL-TR-2020-00171_R1.pdf.

Moorman, R. 2008. "A Safety Re-evaluation of the AVR Pebble Bed Reactor Operation and Its Consequences for Future HTR Concepts." Berichte des Forschungszentrum Jülich 4275. https://juser.fz-juelich.de/record/1304/files/Juel_4275_Moormann.pdf?version=1.

Morris, E. E., and T. H. Bauer. 2005. "Modeling of the Repository Behavior of TRISO Fuel (Milestone 4295 for Work Package A0403K11: Rev.1)." Argonne, IL: Argonne National Laboratory, September 29. https://publications.anl.gov/anlpubs/2006/01/55330.pdf.

Mulder, E. J. 2021. "Overview of X-Energy's 200 MWth Xe-100 Reactor." Presentation to the National Academy of Sciences Committee on Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Reactors, January 13. https://www.nationalacademies.org/event/01-11-2021/merits-and-viability-of-different-nuclear-fuel-cycles-and-technology-options-and-the-waste-aspects-of-advanced-nuclear-january-11-2021-and-january-13-2021-meeting.

Murray, R. L., and K. E. Holbert. 2014. Nuclear Energy: An Introduction to the Concepts, Systems, and Applications of Nuclear Processes. 7th ed. Boston. MA: Butterworth-Heinemann.

Murray, R. L., and K. E. Holbert. 2020. "Nuclear Power Plants." In Nuclear Energy: An Introduction to the Concepts, Systems, and Applications of Nuclear Processes, 8th Edition, pp. 325-348. Cambridge, MA: Elsevier. https://www.sciencedirect.com/science/article/pii/B9780128128817000186?via%3Dihub.

Nabielek, H., H. van der Merwe, J. Fachinger, K. Verfondern, W. von Lensa, B. Grambow, and E. de Visser-Tynova. 2009. "Ceramic Coated Particles for Safe Operation in HTRs and in Long-Term Storage." In Ceramic Engineering and Science Proceedings 30:193-202. Daytona Beach, FL. https://doi.org/10.1002/9780470584002.ch17.

Nagley, S. 2020. "BWXT HA-LEU Supply (Enrichment, De-Conversion, Fuel Fab, and Transportation)." Presented at the HALEU Webinar, April 28. https://gain.inl.gov/HALEU_Webinar_Presentations/12-Nagley,BWXT-28Apr2020.pdf.

Nagy, K., J. L. Kloosterman, D. Lathouwers, and T. H. J. J. van der Hagen. 2008. "Parametric Studies on the Fuel Salt Compositions in Thermal Molten Salt Breeder Reactors." International Conference on the Physics of Reactors, Interlaken, Switzerland, September 14-19.

Napier, B. A. 2020. "Non-LWR Fuel Cycle Environmental Data." PNNL29367 Rev. 1. Richland, WA: Pacific Northwest National Laboratory. March. https://www.nrc.gov/docs/ML2007/ML20076G075.pdf.

NAS (National Academy of Sciences). 1994. Management and Disposition of Excess Weapons Plutonium. Washington, DC: National Academy Press. https://doi.org/10.17226/2345.

NASEM (National Academies of Sciences, Engineering, and Medicine). 2020. Review of the Department of Energy's Plans for Disposal of Surplus Plutonium in the Waste Isolation Pilot Plant. Washington, DC: The National Academies Press. https://doi.org/10.17226/25593.

Nash, K. L., and G. J. Lumetta. 2011. Advanced Separation Techniques for Nuclear Fuel Reprocessing and Radioactive Waste Treatment. Oxford, UK: Woodhouse Publishing Ltd.

National Research Council. 1996. Nuclear Wastes: Technologies for Separations and Transmutation. Washington, DC: National Academy Press. https://doi.org/10.17226/4912.

National Research Council. 2000. Electrometallurgical Techniques for DOE Spent Fuel Treatment. Washington, DC: National Academy Press. https://doi.org/10.17226/9883.

National Research Council. 2005. Monitoring Nuclear Weapons and Nuclear-Explosive Materials: An Assessment of Methods and Capabilities. Washington, DC: The National Academies Press. https://doi.org/10.17226/11265.

National Research Council. 2008. Review of DOE's Nuclear Energy Research and Development Program. Washington, DC: The National Academies Press. https://doi.org/10.17226/11998.

National Research Council. 2012. Analysis of Cancer Risks in Populations Near Nuclear Facilities: Phase 1. Washington, DC: The National Academies Press. https://doi.org/10.17226/13388.

NEA and IAEA (Nuclear Energy Agency and International Atomic Energy Agency). 2020. "Uranium 2020: Resources, Production, and Demand." Nuclear Energy Agency, 2020. https://www.oecd-nea.org/jcms/pl_52718/uranium-2020-resources-production-and-demand.

NEA-OECD (Nuclear Energy Agency–Organisation for Economic Co-operation and Development). 1994. "The Economics of the Nuclear Fuel Cycle." Paris, France: Nuclear Energy Agency. https://www.oecd-nea.org/upload/docs/application/pdf/2020-07/the_economics_of_the_nuclear_fuel_cycle_1994.pdf.

NEA-OECD. 2002. "Accelerator-driven Systems (ADS) and Fast Reactors (FR)." In Advanced Nuclear Fuel Cycles, A Comparative Study." Paris, France. Nuclear Energy Agency. https://www.oecd-nea.org/ndd/reports/2002/nea3109-ads.pdf.

NEA-OECD. 2005. "The Safety of the Nuclear Fuel Cycle." NEA No. 3588. Nuclear Safety. Paris, France: Nuclear Energy Agency. https://doi.org/10.1787/9789264014220-en.

NEA-OECD. 2006a. "Forty Years of Uranium Resources, Production, and Demand in Perspective: 'The Red Book Retrospective.'" NEA No. 6096. Paris, France: Organisation for Economic Co-Operation and Development. https://www.oecd-nea.org/jcms/pl_14100/forty-years-of-uranium-resources-production-and-demand-in-perspective?details=true.

NEA-OECD. 2006b. "Advanced Nuclear Fuel Cycles and Radioactive Waste Management." Nuclear Development. Paris, France: Nuclear Energy Agency, 2006. https://www.oecd-nea.org/upload/docs/application/pdf/2019-12/5990-advanced-nfc-rwm.pdf.

NEA-OECD. 2012. "Homogeneous versus Heterogeneous Recycling of Transuranics in Fast Nuclear Reactors." NEA No. 7077. Nuclear Science. Paris, France: Organisation for Economic Co-Operation and Development, 2012. https://www.oecd-nea.org/jcms/pl_14790/homogeneous-versus-heterogeneous-recycling-of-transuranics-in-fast-nuclear-reactors?details=true.

NEA-OECD. 2013. "The Economics of the Back End of the Nuclear Fuel Cycle." Nuclear Development. Boulogne-Billancourt, France: Nuclear Energy Agency, 2013. https://www.oecd-nea.org/upload/docs/application/pdf/2019-12/7061-ebenfc.pdf.

NEA-OECD. 2015a. "Introduction of Thorium in the Nuclear Fuel Cycle: Short- to Long-Term Considerations." NEA No. 7224. Nuclear Science. Paris: Nuclear Energy Agency. https://www.oecd-nea.org/upload/docs/application/pdf/2019-12/7224-thorium.pdf.

NEA-OECD. 2015b. "Perspectives on the Use of Thorium in the Nuclear Fuel Cycle: Extended Summary." NEA No. 7728. Nuclear Science. Paris, France: Organisation for Economic Co-Operation and Development. https://www.oecd-nea.org/upload/docs/application/pdf/2019-12/7228-thorium-es.pdf.

NEA-OECD. 2018a. Tests and Research Reactor Capabilities for Nuclear Fuel and Material Studies. Background Paper for the Workshop "Building Multinational Fuel and Materials Testing Capacities for Science, Safety and Industry." OECD Conference Center, Boulogne-Billancourt. 4-5 October. https://www.oecd-nea.org/science/workshops/fuelmat2018/documents/BackgroundPaperWorkshopOct4-528_09_2018.pdf.

NEA-OECD. 2018b. "State-of-the-Art Report on the Progress of Nuclear Fuel Cycle Chemistry." Nuclear Science. Boulogne-Billancourt, France: Nuclear Energy Agency. https://www.oecd-nea.org/jcms/pl_14970/state-of-the-art-report-on-the-progress-of-nuclear-fuel-cycle-chemistry.

NEA-OECD. 2019. "A Code-to-Code Benchmark for High-Temperature Gas-Cooled Reactor Fuel Element Depletion." NEA/NSC/R(2019)1. Paris, France: Nuclear Energy Agency, February 2019. https://www.oecd-nea.org/science/wprs/egrpans/documents/nsc-r2019-1.pdf.

NEA-OECD. 2020. "Management and Disposal of High-Level Radioactive Waste: Global Progress and Solutions." NEA No. 7532. Radioactive Waste Management. Paris, France: Nuclear Energy Agency. https://www.oecd-nea.org/rwm/pubs/2020/7532-DGR.pdf.

NEA-OECD. 2021. "Strategies and Considerations for the Back End of the Fuel Cycle." Nuclear Technology Development and Economics. Boulogne-Billancourt, France: Nuclear Energy Agency. https://www.oecd-nea.org/jcms/pl_55928/strategies-and-considerations-for-the-back-end-of-the-fuel-cycle?details=true.

NEA-OECD. n.d. "Nuclear Innovation 2050 (NI2050)." Nuclear Energy Agency. https://www.oecd-nea.org/jcms/pl_21829/nuclear-innovation-2050-ni2050. Accessed March 14, 2022.

NEI (Nuclear Engineering International). 2017. "Russia Looks to Develop Environmentally Friendly Reprocessing." Nuclear Engineering International. November 29. https://www.neimagazine.com/news/newsrussia-looks-to-develop-environmentally-friendly-reprocessing-5989047.

NEI. 2020a. "US and Russia Sign Final Amendment to Uranium Suspension Agreement." Nuclear Engineering International. October 8. https://www.neimagazine.com/news/newsus-and-russia-sign-final-amendment-to-uranium-suspension-agreement-8172402.

NEI. 2020b. "Japan's Rokkasho Reprocessing Plant Postponed Again." Nuclear Engineering International. August 27. https://www.neimagazine.com/news/newsjapans-rokkasho-reprocessing-plant-postponed-again-8105722.

NEI. 2021a. "Terrestrial Upgrades Molten Salt Reactor Design." September 16. https://www.neimagazine.com/news/newsterrestrial-upgrades-molten-salt-reactor-design-9086577.

NEI. 2021b. "First Remix Fuel Assemblies Loaded at Russia's Balakovo Nuclear Plant." Nuclear Engineering International. December 23. https://www.neimagazine.com/news/newsfirst-remix-fuel-assemblies-loaded-at-russias-balakovo-nuclear-plant-9343328.

NEI. 2022. "Vogtle NPP Faces More Possible Delays." Nuclear Engineering International. August 3, 2022. https://www.neimagazine.com/news/newsvogtle-npp-faces-more-possible-delays-9897168.

Neider, T. 2021. "Natrium." Presented at the National Academies' Committee on Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Reactors, February 22. https://www.nationalacademies.org/event/02-22-2021/merits-and-viability-of-different-nuclear-fuel-cycles-and-technology-options-and-the-waste-aspects-of-advanced-nuclear-reactors-february-22-and-23-2021-meeting.

Newton, F., and N. Kolosovskaya. 2020. "HALEU: Rosatom View." Presented at the HALEU Webinar, April 28. https://gain.inl.gov/HALEU_Webinar_Presentations/14-15-Newton-Kolosovskaya,TENEX-28Apr2020.pdf.

Nichol, M. 2021. Current Status of Advanced Nuclear Reactors. Presentation given to the NC Energy Policy Council. August 18, 2021. https://files.nc.gov/ncdeq/energy-policy-council/August_18_2021_EPC_Meeting.pdf

NPT (Treaty on the Non-Proliferation of Nuclear Weapons). 1968. https://www.state.gov/wp-content/uploads/2019/03/228-Nuclear-Nonproliferation-Treaty-508.pdf.

NRC (National Research Council). 1957. The Disposal of Radioactive Waste on Land. Washington, DC: National Academy Press. https://doi.org/10.17226/10294.

NSUF (Nuclear Science User Facilities). n.d. "NSUF Facilities and Foreign Affiliates." Nuclear Science User Facilities. https://nsuf.inl.gov/Home/PartnerFacilities. Accessed July 29, 2022.

Nuclear Decommissioning Collaborative. 2022. "Trojan." The Nation's Decommissioning Clearinghouse. 2022. https://decommissioningcollaborative.org/trojan/.

Nuclear Energy Institute. 2018. "Addressing the Challenges with Establishing the Infrastructure for the Front-End of the Nuclear Fuel Cycle for Advanced Reactors." Washington, DC: Nuclear Energy Institute. January. https://www.nrc.gov/docs/ML1810/ML18103A250.pdf.

NuclearSAFE. 2018. "Technology." https://nstusa.net/technology.

NuScale Power. 2020. "NuScale Final Safety Analysis Report, Chapter 4: Reactor." ML20224A492. NuScale Standard Plant Design Certification Application. NuScale Power. https://www.nrc.gov/docs/ML2022/ML20224A492.pdf.

NuScale Power. 2022a. "Overview of Design Optimization Process for a Commercial Nuclear Core." NuScale Power, August 26, 2022.

NuScale Power. 2022b. NuScale Response to the National Academies of Sciences, Engineering, and Medicine Questionnaire, NuScale Power, WP-120652, Rev. A, June 22, 2022.

Nutt, M. 2021. "Advanced Non-Light Water Reactors: Integrated Waste Management System Considerations." Presented at the National Academies' Committee on Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Reactors, online, December 16. https://www.nationalacademies.org/event/12-16-2021/merits-and-viability-of-different-nuclear-fuel-cycles-and-technology-options-and-the-waste-aspects-of-advanced-nuclear-reactors-meeting-13-december-16-2021.

NWTRB (Nuclear Waste Technical Review Board). 2015. "Designing a Process for Selecting a Site for a Deep-Mined, Geologic Repository for High-Level Radioactive Waste and Spent Nuclear Fuel: A Detailed Analysis." Report to the U.S. Congress and Secretary of Energy. Arlington, VA: U.S. Nuclear Waste Technical Review Board, November 2015. https://www.nwtrb.gov/docs/default-source/reports/siting_report_analysis.pdf?sfvrsn=9.

NWTRB. 2017. "Management and Disposal of U.S. Department of Energy Spent Nuclear Fuel: A Report to the United States Congress and the Secretary of Energy." Arlington, VA: U.S. Nuclear Waste Technical Review Board. December. https://www.nwtrb.gov/docs/default-source/reports/nwtrb-mngmntanddisposal-dec2017-508a.pdf?sfvrsn=12.

NWTRB. 2020. "Department of Energy-Managed Spent Nuclear Fuel at the Hanford Site." U.S. Nuclear Waste Technical Review Board. June. https://www.nwtrb.gov/docs/default-source/facts-sheets/doe-snf-fact-sheet---hanford-rev-1.pdf?sfvrsn=8.

Nygaard, E. 2021. "BWXT's Advanced Nuclear Reactor (BANR)." Presentation to the National Academies' Committee on Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Reactors. January 11. https://www.nationalacademies.org/event/01-11-2021/merits-and-viability-of-different-nuclear-fuel-cycles-and-technology-options-and-the-waste-aspects-of-advanced-nuclear-january-11-2021-and-january-13-2021-meeting.

Okada, K. 1985. "Separation Method for a Spent Fuel Rod." European Patent, Application No. 19850302514.

Oklo. 2020. "Combined Operating License Application, Part V: Non-Applicabilities and Requested Exemptions." OkloPower-2020-PartV-NP, Rev. 0. Oklo, Inc. https://www.nrc.gov/docs/ML2007/ML20075A006.pdf.

Olson, G. L., R. K. McCardell, and D. B. Illum. 2002. "Fuel Summary Report: Shippingport Light Water Breeder Reactor—Rev. 2." INEEL/EXT-98-00799. Idaho Falls, ID: Idaho National Engineering and Environmental Laboratory. September 1. https://doi.org/10.2172/910649.

Orano. 2021a. "MOX, Recycling Nuclear Energy." https://www.orano.group/country/china/en/our-stories/mox-recycling-nuclear-energy.

Orano. 2021b. "Orano's activity in China." https://www.orano.group/en/orano-across-the-world/china#OranoactivityinChina. Accessed August 5, 2021.

Orano. n.d. "Orano La Hague." https://www.orano.group/country/china/en/our-stories/orano-la-hague. Accessed February 22, 2022.

Oregon DOE (Department of Energy). n.d. "Trojan Nuclear Site Spent Fuel Storage." Oregon.gov. https://www.oregon.gov/energy/safety-resiliency/pages/trojan-site.aspx. Accessed December 1, 2021.

O'Sullivan, R. 2021. "Moltex Technology." Presentation to the National Academies' Committee on Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Reactors, February 23, 2021. https://www.nationalacademies.org/event/02-22-2021/merits-and-viability-of-different-nuclear-fuel-cycles-and-technology-options-and-the-waste-aspects-of-advanced-nuclear-reactors-february-22-and-23-2021-meeting.

OTA (Office of Technology Assessment). 1977. "Nuclear Proliferation and Safeguards." NTIS order #PB-275843. Washington, DC: Office of Technology Assessment, June 1977. https://ota.fas.org/reports/7705.pdf.

Pappano, P. 2021. "X-Energy TRISO Fuel Production." Presented at the National Academies' Committee on the Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Nuclear Reactors, February 23.

Patel, S. 2019. "France Scraps Fast Nuclear Reactor Demonstration." POWER, November 1, 2019. https://www.powermag.com/france-scraps-fast-nuclear-reactor-demonstration.

Patel, S. 2021. "Honeywell to Reopen Sole U.S. Uranium Conversion Plant." POWERnews, February 9. https://www.powermag.com/honeywell-to-reopen-sole-u-s-uranium-conversion-plant.

Patel, S. 2022. "Nuclear Field Activities Completed for Idaho NuScale SMR Project." POWERnews, February 3. https://www.powermag.com/nuclear-field-activities-completed-for-idaho-nuscale-smr-project.

Patterson, M. N. 2021. "Update on EBR-II Used Fuel Treatment." INL/EXT-21-64586. Presentation to the National Academies' Committee on Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Reactors, September 29, 2021. https://www.nationalacademies.org/event/09-28-2021/merits-and-viability-of-different-nuclear-fuel-cycles-and-technology-options-and-the-waste-aspects-of-advanced-nuclear-reactors-meeting-10-september-28-29-2021-public-sessions.

Paunov, P., and I. Naydenov. 2020. "Long-Term Radiotoxicity Evaluation of PWR Spent Uranium and MOX Fuel and Highly Active Waste." E3S Web of Conferences. 207. https://doi.org/10.1051/e3sconf/202020701024.

Paviet-Hartmann, P., G. Cerefice, M. Stacey, and S. Bakhtiar. 2011. "Analysis of Nuclear Proliferation Resistance Reprocessing and Recycling Technologies." In Report Number: INL/CON-10-20124. Makuhari, Japan. https://www.osti.gov/biblio/1023498.

Pereira, C. 2020. "Molten Salt Reactor Recycle Option." Presented at the Committee on Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Reactors, December 8. https://www.nationalacademies.org/event/12-07-2020/merits-and-viability-of-different-nuclear-fuel-cycles-and-technology-options-and-the-waste-aspects-of-advanced-nuclear-december-7-8-2020-meeting.

Peretz, F. J., J. E. Rushton, R. L. Faulkner, K. L. Walker, and G. D. Del Cul. 1998. "Removal of Uranium and Salt from the Molten Salt Reactor Experiment." ORNL/CP-98146. Oak Ridge, TN: Oak Ridge National Laboratory.

Perry, R. 2019. "Energy Secretary's Report to Congress Regarding Atomic Energy Defense Act (50 U.S.C. Section 2566)." April 23. https://nuke.fas.org/guide/usa/mox-rept-2019.pdf.

Peters, M. T., and R. C. Ewing. 2007. "A Science-Based Approach to Understanding Waste Form Durability in Open and Closed Nuclear Fuel Cycles." E-MRS 2006: Symposium N 362 (2, May 31):395-401. https://doi.org/10.1016/j.jnucmat.2007.01.085.

Peters, S., D. Vinson, and J. T. Carter. 2020. "Spent Nuclear Fuel and Reprocessing Waste Inventory." U.S. Department of Energy, September 2020. https://sti.srs.gov/fulltext/FCRD-NFST-2013-000263_R7.pdf.

Peterson, J., and M. L. Dunzik-Gougar. 2011. "Modeling Spent TRISO Fuel for Geological Disposal: Corrosion and Failure Under Oxidizing Conditions in the Presence of Water." Progress in Nuclear Energy 53(3):278-284. https://doi.org/10.1016/j.pnucene.2010.12.003.

Peterson, J., and J. Wagner. 2014. "Characteristics of Commercial Spent Nuclear Fuel: Distributed, Diverse, and Changing with Time." Radwaste Solutions January-March:50-59.

Petti, D., R. Hill, J. Gehin, H. Gougar, G. Strydom, F. Heidet, J. Kinsey, C. Grandy, A. Qualls, N. Brown, J. Powers, E. Hoffman, and D. Croson. 2017. "Advanced Demonstration and Test Reactor Options Study." Idaho Falls, ID: Idaho National Laboratory. January.

Pierce, R. A. 2017. Low temperature chemical processing of graphite-clad nuclear fuels. 9,793,019. Aiken, SC, issued October 17. https://patft.uspto.gov/netacgi/nph-Parser?Sect1=PTO1&Sect2=HITOFF&d=PALL&p=1&u=%2Fnetahtml%2FPTO%2Fsrchnum.htm&r=1&f=G&l=50&s1=9793019.PN.&OS=PN/9793019&RS=PN/9793019.

Piet, S. J. 2013. "When Is the Simple Radiotoxicity Approach Useful for Advanced Fuel Cycle System Assessments Given the Existence of Complex Performance Dose Assessments?" Nuclear Science and Engineering 173(1, January 1):58-81. https://doi.org/10.13182/NSE11-114.

Podvig, P. 2021. "Russia's Efforts to Develop Advanced Nuclear Cycles." Presented at the Committee on Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Reactors, September 15. https://www.nationalacademies.org/event/09-13-2021/merits-and-viability-of-different-nuclear-fuel-cycles-and-technology-options-and-the-waste-aspects-of-advanced-nuclear-reactors-meeting-9-september-13-15-2021.

Poinssot, C. 2021. "What Are the Advantages and Drawbacks of Recycling Spent Nuclear Fuels, and Why Recycling Can Be a Significant Contribution to Sustainable Nuclear Energy Systems?" White Paper for National Academies' Committee on the Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Nuclear Reactors, 2021.

Poinssot, C., and B. Boullis. 2012. "Actinide Recycling within Closed Fuel Cycles." Nuclear Engineering International, February 14. https://www.neimagazine.com/features/featureactinide-recycling-within-closed-fuel-cycles.

Poinssot, C., B. Boullis, J.-M. Adnet, and E. Touron. 2017a. From the Bataille's Law to the Am-recycling integral experiments, overview of 25 years of R&D in CEA on Minor Actinides Partitioning Processes. GLOBAL 2017, September, Seoul, South Korea. https://hal.science/cea-02434525.

Poinssot, C., S. Grandjean, S. Bourg, J. Serp, E. Touron, and B. Boullis. 2017b. French Perspective on the Respective Interests of Hydro- and Pyro-Chemical Processes for Recycling Future Spent Nuclear Fuels. GLOBAL 2017, September, Seoul, South Korea. https://hal.science/cea-02434536.

Poneman, D., and Cutlip, L. 2021. "Status and Prospects for HALEU Production in the United States." Presentation to the National Academies' Committee on Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Reactors, June 2. https://www.nationalacademies.org/event/06-01-2021/merits-and-viability-of-different-nuclear-fuel-cycles-and-technology-options-and-the-waste-aspects-of-advanced-nuclear-reactors-meeting-6-part-1-june-1-2-2021.

Postovarova, D. V., N. V. Kovalev, M. S. Onegin, and B. A. Bibichev. 2016. "Radiation Characteristics of REMIX Fuel During Multiple Recycling in VVER-1000 Reactors." Nuclear Energy and Technology 2(2, June 1):119-125. https://doi.org/10.1016/j.nucet.2016.05.009.

Prabhu, D. R., G. R. Mahajan, and G. M. Nair. 1997. "Di(2-Ethyl Hexyl) Butyramide and Di(2-Ethyl Hexyl)Isobutyramide as Extractants for Uranium(VI) and Plutonium(IV)." Journal of Radioanalytical and Nuclear Chemistry 224(1):113-117. https://doi.org/10.1007/BF02034622.

Ramana, M. V., and E. Saikawa. 2011. "Choosing a Standard Reactor: International Competition and Domestic Politics in Chinese Nuclear Policy." Energy 36(12, December 1):6779-6789. https://doi.org/10.1016/j.energy.2011.10.022.

Reagan, R. 1981. "Statement Announcing a Series of Policy Initiatives on Nuclear Energy." https://www.reaganlibrary.gov/archives/speech/statement-announcing-series-policy-initiatives-nuclear-energy.

Rechard, R. P., T. Hadgu, Y. Wang, L. C. Sanchez, P. McDaniel, C. Skinner, N. Fathi, S. Frank, and M. Patterson. 2017. "Feasibility of Direct Disposal of Salt Waste from Electrochemical Processing of Spent Nuclear Fuel." In International High-Level Radioactive Waste Management. Charlotte, NC. https://www.osti.gov/servlets/purl/1431544.

Recktenwald, G. D., and M. R. Deinert. 2012. "Cost Probability Analysis of Reprocessing Spent Nuclear Fuel in the US." Energy Economics 34(6):1873-1881. https://doi.org/10.1016/j.eneco.2012.07.016.

Redmond, E., and N. Ashkeboussi. 2022. "Establishing a High Assay Low Enriched Uranium Infrastructure for Advanced Reactors." Washington, DC: Nuclear Energy Institute. https://www.nei.org/CorporateSite/media/filefolder/resources/reports-and-briefs/NEI-White-Paper-Establishing-a-High-Assay-Low-Enriched-Uranium-Infrastructure-for-Advanced-Reactors-Jan-2022.pdf.

Regalbuto, M. 2021. "Independent Report: GAIN/EPRI/NEI April 2020 Virtual HALEU Workshop." Presentation to the National Academies' Committee on the Merits and Viability of Different Fuel Cycles and Technology Options and the Waste Aspects of Advanced Nuclear Reactors, September 28, 2021. https://www.nationalacademies.org/event/09-28-2021/merits-and-viability-of-different-nuclear-fuel-cycles-and-technology-options-and-the-waste-aspects-of-advanced-nuclear-reactors-meeting-10-september-28-29-2021-public-sessions.

Regan, C., J. Hammelman, and J. Cuadrado. 2020. "Overview of the NRC's Regulatory Programs on the Advanced Reactor Fuel Cycle." Presented at the National Academies' Committee on Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Reactors, September 22.

Reset Steering Committee. 2018. Reset of America's Nuclear Waste Management: Strategy and Policy. Stanford University Center for International Security and Cooperation and George Washington University Elliott School of International Affairs. October 15. https://fsi-live.s3.us-west-1.amazonaws.com/s3fs-public/reset_report_2018_final.pdf. Accessed April 21, 2022.

Reyes, J. 2021. "NuScale Power—A Scalable Clean Energy Solution." Presentation to the National Academies' Committee on Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Reactors, January 11, 2021. https://www.nationalacademies.org/event/01-11-2021/merits-and-viability-of-different-nuclear-fuel-cycles-and-technology-options-and-the-waste-aspects-of-advanced-nuclear-january-11-2021-and-january-13-2021-meeting.

Rieman, C. R. 2013. "West Valley Demonstration Project: Successes and Challenges." In Proceedings of WM '13 Symposium. Phoenix, AZ.

Riley, B. J., J. McFarlane, G. D. DelCul, J. D. Vienna, C. I. Contescu, L. M. Hay, A. V. Savino, and H. E. Adkins. 2018. "Identification of Potential Waste Processing and Waste Form Options for Molten Salt Reactors." NTRD-MSR-2018-000379, PNNL-27723. Nuclear Technology Research and Development. Pacific Northwest National Laboratory and Oak Ridge National Laboratory, August 15. https://info.ornl.gov/Sites/Publications/Files/Pub114284.pdf.

Riley, B., J. McFarlane, G. Del Cul, J. Vienna, C. Contescu, and C. Forsberg. 2019. "Molten Salt Reactor Waste and Effluent Management Strategies: A Review." Nuclear Engineering and Design 345(April 15):94-109. https://doi.org/10.1016/j.nucengdes.2019.02.002.

Rivers, J., J. Wheeler, and G. Couture. 2021. "Developing A Security Program for the X-Energy HALEU Fuel Fabrication Facility to Support Advanced Reactors." Institute of Nuclear Materials Management 62nd Annual Meeting. https://resources.inmm.org/sites/default/files/2021-09/a1622.pdf.

Robel, M., J. Choi, B. B. Ebbinghaus, B. W. Sleaford, C. G. Bathke, B. A. Collins, Z. Beauvais, K. R. Hase, A. W. Prichard, and J. A. Blink. 2013. "A Safeguards and Security Assessment Comparing the Nuclear Material Attractiveness of Unirradiated and Irradiated Fuels Associated with Existing Power Reactors and Potential Future Small Modular Reactors." https://www.osti.gov/servlets/purl/1088489.

Rodrigues, D., G. Durán-Klie, and S. Delpech. 2015. "Pyrochemical Reprocessing of Molten Salt Fast Reactor Fuel: Focus on the Reductive Extraction Step." Nukleonika 60(4):907-914. https://doi.org/doi:10.1515/nuka-2015-0153.

Rodríguez-Penalonga, L., and B. Y. Moratilla Soria. 2017. "A Review of the Nuclear Fuel Cycle Strategies and the Spent Nuclear Fuel Management Technologies." Energies 10(8). https://doi.org/10.3390/en10081235.

Rothwell, G., Wood, T., Daly, D. and Weimar, M. 2014. "Sustainability of Light Water Reactor Fuel Cycles." Nuclear Energy and Sustainable Development: Selected Topics 74(December):S16-S23. https://doi.org/10.1016/j.enpol.2014.07.018.

Sackett, J., and E. D. Arthur. 2021. "The ARC-100 Advanced SMR." Presentation to the National Academies' Committee on Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Reactors, February 22. https://www.nationalacademies.org/event/02-22-2021/merits-and-viability-of-different-nuclear-fuel-cycles-and-technology-options-and-the-waste-aspects-of-advanced-nuclear-reactors-february-22-and-23-2021-meeting.

Scheinman, L. 1987. The International Atomic Energy Agency and World Nuclear Order. New York: Routledge.

Schneider, E. A., M. R. Deinert, and K. B. Cady. 2009. "Cost Analysis of the US Spent Nuclear Fuel Reprocessing Facility." Energy Economics 31(5):627-634. https://doi.org/10.1016/j.eneco.2008.12.011.

SCK-CEN. 2018. "BR2 Belgian Reactor 2 US National Scientific User Affiliate Facility User's Guide." Belgium: SCK-CEN. January.

Scott, L. M., M. P. Dion, K. K. Hogue, S. E. O'Brien, and G. T. Westphal. 2021. "Infrastructure and Testing Needs for Molten Salt Reactor Safeguards." ORNL/TM-2021/1940. Oak Ridge, TN: Oak Ridge National Laboratory. March 1. https://doi.org/10.2172/1782030.

Selectra. 2021. "Electricity Tariffs in France: Compare Electricity Prices in 2021." Paris, France: Selectra. https://en.selectra.info/energy-france/guides/electricity/tariffs.

Serco. 2011. "Literature Review of Partitioning and Transmutation." Prepared for Nuclear Decommissioning Authority (NDA) Radioactive Waste Management Directorate (RWMD). United Kingdom: Serco. February. https://webarchive.nationalarchives.gov.uk/ukgwa/20211004151120/https://rwm.nda.gov.uk/publication/literature-review-of-partitioning-and-transmutation.

Shafer, J. 2021. "Advanced Research Project Agency-Energy (ARPA-E) Efforts Supporting Advanced Nuclear." Presented at the Committee on Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Reactors, September 13. https://www.nationalacademies.org/event/09-13-2021/merits-and-viability-of-different-nuclear-fuel-cycles-and-technology-options-and-the-waste-aspects-of-advanced-nuclear-reactors-meeting-9-september-13-15-2021.

Sherman, S. R., and C. J. Knight. 2005. "Treatment Method for Fermi Barrel Sodium Metal Residues." INL/EXT-05-00421. Idaho Falls, ID: Idaho National Laboratory. June 1. https://doi.org/10.2172/912436.

Shoesmith, D. W. 2000. "Fuel Corrosion Processes Under Waste Disposal Conditions." Journal of Nuclear Materials 282(1, November 1):1-31. https://doi.org/10.1016/S0022-3115(00)00392-5.

Shoman, N., and M. Higgins. 2021. "Considerations for NMA of Molten Salt Reactors," American Nuclear Society Winter Meeting, Washington, DC. November.

Shropshire, D., G. Black, and K. Araujo. 2021. "Global Market Analysis of Microreactors." DOE Microreactor Program. Idaho Falls, ID: Idaho National Laboratory. https://gain.inl.gov/MicroreactorProgramTechnicalReports/Document-INL-EXT-21-63214.pdf.

Sivagami, K., K. Padmanabhan, A. C. Joy, and I. M. Nambi. 2019. "Microwave (MW) Remediation of Hydrocarbon Contaminated Soil Using Spent Graphite—An Approach for Waste as a Resource." Journal of Environmental Management 230(January 15):151-158. https://doi.org/10.1016/j.jenvman.2018.08.071.

SNETP (Sustainable Nuclear Energy Technology Platform). 2011. "Thorium Cycles and Thorium as a Nuclear Fuel Component." Strategic Research Agenda—Annex. Brussels, Belgium: Sustainable Nuclear Energy Technology Platform. January. Thorium Cycles and Thorium as a Nuclear Fuel Component.

Soares, A., A. Lintereur, B. R. Betzler, A. Johnsen, J. Flygare, L. G. Worrall, and W. Walters. 2020. "Nuclear Data Uncertainty Challenges in Molten Salt Reactor Safeguards." In Oak Ridge, TN. https://www.osti.gov/biblio/1763462.

Sorensen, K. 2021. "Liquid-Fluoride Thorium Reactor Design and Fuel Cycle." Presentation to the National Academies' Committee on Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Reactors, January 13.

SRS (Savannah River Site). 2020. "Defense Waste Processing Facility." December. https://www.srs.gov/general/news/factsheets/srr_dwpf.pdf.

Stamp, E. 2020. "This Sleek Building Is Actually a Nuclear Reactor." Architectural Digest, March 23. https://www.architecturaldigest.com/story/sleek-building-actually-nuclear-reactor.

Stern, W. 2021. "IAEA Safeguards Considerations Associated with HALEU." Presentation to the National Academies' Committee on Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Reactors, May 17. https://www.nationalacademies.org/event/05-17-2021/merits-and-viability-of-different-nuclear-fuel-cycles-and-technology-options-and-the-waste-aspects-of-advanced-nuclear-reactors-may-17-19-2021-meeting.

Subsidy Tracker. 2021. "NuScale Power." https://subsidytracker.goodjobsfirst.org/prog.php?parent=nuscale-power&order=sub_year&sort=asc. Accessed December 30, 2021.

Suk, P., O. Chvála, G. I. Maldonado, and J. Frýbort. 2021. "Simulation of a NuScale Core Design with the CASL VERA Code." Nuclear Engineering and Design 371(January):110956. https://doi.org/10.1016/j.nucengdes.2020.110956.

Swift, P. 2021. "Impacts of Nuclear Fuel Cycle Choices on Permanent Disposal of High-Activity Radioactive Waste." Presented at the Committee on Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Reactors, June 2. https://www.nationalacademies.org/event/06-01-2021/merits-and-viability-of-different-nuclear-fuel-cycles-and-technology-options-and-the-waste-aspects-of-advanced-nuclear-reactors-meeting-6-part-1-june-1-2-2021.

Swift, P. N., and D. Sassani. 2019. "Impacts of Nuclear Fuel Cycle Choices on Permanent Disposal of High-Activity Radioactive Wastes." SAND2019-5941C, 675860. https://www.osti.gov/biblio/1640197.

Szieberth, M., M. Halász, T. Reiss, and S. Fehér. 2013. "Fuel Cycle Studies on Minor Actinide Burning in Gas Cooled Fast Reactors." In Proceedings of an International Conference Held in Paris, France, Track 8-Deployment and Scenarios. Paris, France: International Atomic Energy Agency. https://www-pub.iaea.org/MTCD/Publications/PDF/SupplementaryMaterials/P1665CD/Track8_Deployment_and_Scenarios.pdf.

Szondy, D. 2018. "Cost-Effective Method of Extracting Uranium from Seawater Promises Limitless Nuclear Power." New Atlas, June 14, 2018. https://newatlas.com/nuclear-uranium-seawater-fibers/55033.

Tan, Y-Z., Z-Y. Xie, F. Peng, F-H. Qian, and H. J. Ming. 2021. "Optimal Mixing Scheme for Graphite–Bentonite Mixtures Used as Buffer Materials in High-Level Waste Repositories." Environmental Earth Sciences 80(17):549. https://doi.org/10.1007/s12665-021-09809-x.

Taylor, R. (Ed.). 2015. Reprocessing and Recycling of Spent Nuclear Fuel. Woodhead Publishing Series in Energy. Oxford, UK: Woodhead Publishing. https://doi.org/10.1016/C2013-0-16483-5.

TerraPower. 2021a. The Natrium™ Program. May 18. https://www.terrapower.com/natrium-program-summary. Accessed April 21, 2022.

TerraPower. 2021b. "Additional TerraPower Information for Review Committees." August.

TerraPower. 2022. "Additional Information Provided in Response to Fact Checking." June 2.

Terrestrial Energy. 2021. "Response to the National Academy of Sciences Questionnaire on IMSR® Technology and Development." February.

The Economist. 2022. "The First Underground Warren for Disposing of Spent Nuclear Fuel," June 23, 2022. https://www.economist.com/science-and-technology/2022/06/22/the-first-underground-warren-for-disposing-of-spent-nuclear-fuel.

Todd, T. 2020. "Ceramic and TRISO Fuel Recycle Option." Presented to the National Academies' Committee on Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Reactors, December 8. https://www.nationalacademies.org/event/12-07-2020/merits-and-viability-of-different-nuclear-fuel-cycles-and-technology-options-and-the-waste-aspects-of-advanced-nuclear-december-7-8-2020-meeting.

Todd, T. 2021. "Trends in Nuclear Fuel Cycle Workforce." Presented at the Committee on Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Reactors, December 6.

Toki, M., and M. Pomper. 2013. "Time to Stop Reprocessing in Japan." Arms Control Today 43(1, January/February):22-29.

Totemeier, A. 2021. "Lightbridge Fuel™ Overview." Presented at the Spring 2021 Nuclear Waste Technical Review Board Meeting, online, May 12. https://www.nwtrb.gov/docs/default-source/meetings/2021/may/totemeier.pdf?sfvrsn=4.

Touran, N. 2020. "Myths and Misconceptions About Thorium Nuclear Fuel." What Is Nuclear, Updated September 8. https://whatisnuclear.com/thorium-myths.html.

Tracy, C. L. M. K. Dustin and R. C. Ewing. 2016. Policy: Reassess New Mexico's nuclear-waste repository. Nature 529(January 13):149-151. https://doi.org/10.1038/529149a.

Triplett, B. S., E. P. Loewen, and B. J. Dooies. 2012. "PRISM: A Competitive Small Modular Sodium-Cooled Reactor." Nuclear Technology 178(May):186-200.

Tuček, K., J. Carlsson, D. Vidović, and H. Wider. 2008. "Comparative Study of Minor Actinide Transmutation in Sodium and Lead-Cooled Fast Reactor Cores." Progress in Nuclear Energy 50(2):382-388. March 1. https://doi.org/10.1016/j.pnucene.2007.11.021.

Tylka, M. M., J. L. Willit, J. Prakash, and M. A. Williamson. 2015. "Application of Voltammetry for Quantitative Analysis of Actinides in Molten Salts." Journal of the Electrochemical Society 162(12):H852-H859. https://doi.org/10.1149/2.0281512jes.

Uhlíř, J., and M. Marecek. 2009. "Fluoride Volatility Method for Reprocessing of LWR and FR Fuels." Journal of Fluorine Chemistry 130(1):89-93. https://doi.org/10.1016/j.jfluchem.2008.07.002.

UN (United Nations) Office of Disarmament Affairs. n.d.-a. "Treaty on the Non-Proliferation of Nuclear Weapons (NPT)." United Nations. https://www.un.org/disarmament/wmd/nuclear/npt. Accessed April 21, 2022.

UN Office of Disarmament Affairs. n.d.-b. Disarmament Treaties Database. https://treaties.unoda.org. Accessed April 21, 2022.

U.S. NRC (U.S. Nuclear Regulatory Commission). 1978. Federal Register, Notices, 43(101). May 24, 1978. https://www.govinfo.gov/content/pkg/FR-1978-05-24/pdf/FR-1978-05-24.pdf.

U.S. NRC. 1983. "Standard Format and Content for a Licensee Physical Security Plan for the Protection of Special Nuclear Material of Moderate or Low Strategic Significance." NRC Regulatory Guide 5.59. Washington, DC: U.S. Nuclear Regulatory Commission, February 1983. https://www.nrc.gov/docs/ML1003/ML100341301.pdf.

U.S. NRC. 2000. "NRC Review of the Tokai-Mura Criticality Accident." Washington, DC: U.S. Nuclear Regulatory Commission. https://www.nrc.gov/reading-rm/doc-collections/commission/secys/2000/secy2000-0085/attachment1.pdf.

U.S. NRC. 2005. "Regulatory Issue Summary 2005-22: Requirements for the Physical Protection During Transportation of Special Nuclear Material of Moderate and Low Strategic Significance." 10 CFR Part 73 vs. Regulatory Guide 5.59 (1983). October 28. ML052000423. https://www.nrc.gov/reading-rm/doc-collections/gen-comm/reg-issues/2005/ri200522.pdf.

U.S. NRC. 2007. Licenses, certifications and approvals for nuclear power plants; Final rule. Federal Register 72(166):49437.

U.S. NRC. 2009. "Material Categorization and Future Fuel Cycle Facility Security-Related Rulemaking." SECY-09-0123. Washington, DC: U.S. Nuclear Regulatory Commission, September 4, 2009. https://www.nrc.gov/docs/ML1228/ML12285A057.pdf.

U.S. NRC. 2010. Waste Confidence Decision Update: Final Rule [10 CFR Part 51, 7590-01-P, NRC-2008-0482]. December 9. Rockville, MD: U.S. Nuclear Regulatory Commission. https://www.nrc.gov/docs/ML1033/ML103350186.pdf. Accessed March 14, 2022.

U.S. NRC. 2011. "Draft Regulatory Basis for Licensing and Regulating Reprocessing Facilities." SECY-11-0163. Washington, DC: U.S. Nuclear Regulatory Commission, November 2011. https://www.nrc.gov/docs/ML1120/ML112081702.pdf.

U.S. NRC. 2013a. "Implementation Guidance for 10CFR37, Physical Protection of Category I and Category 2 Quantities of Radioactive Materials." NUREG-2155. Washington, DC: U.S. Nuclear Regulatory Commission, February 2013. https://www.nrc.gov/docs/ML1305/ML13053A061.pdf.

U.S. NRC. 2013b. "Reprocessing Regulatory Framework—Status and Next Steps." SECY-13-0093. Washington, DC: U.S. Nuclear Regulatory Commission, August 30, 2013. https://www.nrc.gov/reading-rm/doc-collections/commission/secys/2013/2013-0093scy.pdf.

U.S. NRC. 2014a. "Continued Storage of Spent Nuclear Fuel: Final Rule [NRC-2012-0246, RIN 3150-AJ20]." Federal Register 79(182):56238-56263. September 19. https://www.govinfo.gov/content/pkg/FR-2014-09-19/pdf/2014-22215.pdf.

U.S. NRC. 2014b. "Generic Environmental Impact Statement for the Continued Storage of Spent Nuclear Fuel, Final Report." NUREG-2157. Washington, DC: U.S. Nuclear Regulatory Commission, September 2014. https://www.nrc.gov/docs/ML1419/ML14196A105.pdf.

U.S. NRC. 2014c. "Physical Security Best Practices for the Protection of Risk-Significant Radioactive Material." NUREG-2166. Washington, DC: U.S. Nuclear Regulatory Commission, May 2014. https://www.nrc.gov/docs/ML1415/ML14150A382.pdf.

U.S. NRC. 2014d. "Safety and Regulatory Issues of the Thorium Fuel Cycle." NUREG/CR-7176, ORNL/TM-2013/543. Washington, DC: U.S. Nuclear Regulatory Commission, 2014. https://www.nrc.gov/docs/ML1405/ML14050A083.pdf.

U.S. NRC. 2015. "Rulemaking for Enhanced Security of Special Nuclear Material, Regulatory Basis Document." RIN number: 3150-AJ41; NRC Docket ID: NRC-2014-0118. Regulatory Basis Document. Washington, DC: U.S. Nuclear Regulatory Commission, January 2015. https://www.nrc.gov/docs/ML1432/ML14321A007.pdf.

U.S. NRC. 2016. Report to Congress Under Public Law 113-235: Effectiveness of Part 37 of Title 10 of the Code of Federal Regulations. https://www.nrc.gov/docs/ML1634/ML16347A398.pdf. Accessed April 21, 2022.

U.S. NRC. 2017. "Stages of the Nuclear Fuel Cycle." Nuclear Materials, August 22, 2017. https://www.nrc.gov/materials/fuel-cycle-fac/stages-fuel-cycle.html.

U.S. NRC. 2018a. "Backgrounder on High Burnup Spent Nuclear Fuel." U.S. NRC, October 1, 2018. https://www.nrc.gov/reading-rm/doc-collections/fact-sheets/bg-high-burnup-spent-fuel.html.

U.S. NRC. 2018b. "Options and Recommendation for Physical Security for Advanced Reactors." SRM-SECY-18-0076. Washington, DC: U.S. Nuclear Regulatory Commission, November 19, 2018. https://www.nrc.gov/docs/ML1832/ML18324A478.pdf.

U.S. NRC. 2019a. "Discontinuation of Rulemaking—Enhanced Security of Special Nuclear Material." SECY-19-0095. Policy Issue (Notation Vote). Washington, DC: U.S. Nuclear Regulatory Commission, October 1, 2019. https://www.nrc.gov/docs/ML1915/ML19157A084.pdf.

U.S. NRC. 2019b. "Summary of Views of Some Technical Staff on the Alternate Approach to Conduct a Rulemaking Consistent with the 2015 Final Regulatory Basis." SECY-19-0095: Enclosure 2. Washington, DC: U.S. Nuclear Regulatory Commission, October 17, 2019. https://www.nrc.gov/docs/ML1924/ML19248C233.pdf.

U.S. NRC. 2020a. "Accident Tolerant Fuel Regulatory Activities: Increased Enrichment." U.S. NRC. November 13, 2020. https://www.nrc.gov/reactors/atf/enrichment.html.

U.S. NRC. 2020b. "Power Reactors." Nuclear Reactors, April 29, 2020. https://www.nrc.gov/reactors/power.html.

U.S. NRC. 2020c. "Fuel Fabrication." Nuclear Materials, December 2, 2020. https://www.nrc.gov/materials/fuel-cycle-fac/fuel-fab.html#cat1

U.S. NRC. 2020d. "Standard Review Plan for Transportation Packages for Spent Fuel and Radioactive Material." NUREG-2216. Washington, DC: U.S. Nuclear Regulatory Commission, August 2020. https://www.nrc.gov/docs/ML2023/ML20234A651.pdf.

U.S. NRC. 2020e. "Standard Review Plan for Spent Fuel Dry Storage Systems and Facilities—Final Report." NUREG-2215. Washington, DC: U.S. Nuclear Regulatory Commission, April 2020. https://www.nrc.gov/docs/ML2012/ML20121A190.pdf.

U.S. NRC. 2020f. "Certification of Packages for Transportation of Unirradiated Material—Increased Enrichment." U.S. NRC, September 18, 2020. https://www.nrc.gov/reactors/atf/cert-package-transport-unirradiated.html#increased.
U.S. NRC. 2020g. "Mixed Oxide Fuel Fabrication Facility Licensing." U.S. NRC, December 2, 2020. https://www.nrc.gov/materials/fuel-cycle-fac/mox/licensing.html.
U.S. NRC. 2020h. "Path Forward and Recommendations for Certain Low-Level Radioactive Waste Disposal Rulemakings." SECY-20-0098. Washington, DC: U.S. Nuclear Regulatory Commission. https://www.nrc.gov/docs/ML2014/ML20143A165.pdf.
U.S. NRC. 2020i. "NDAA Section 3116 Waste Incidental to Reprocessing (WIR) at the Idaho National Laboratory (INL) in Idaho—Idaho Nuclear Technical and Engineering Center (INTEC) Tank Farm Facility (TFF)." U.S. Nuclear Regulatory Commission, August 26, 2020. https://www.nrc.gov/waste/incidental-waste/wir-process/wir-locations/idaho-national-lab.html.
U.S. NRC. 2020j. "Consolidated Interim Storage Facility (CISF)." U.S. NRC. December 8, 2020. https://www.nrc.gov/waste/spent-fuel-storage/cis.html.
U.S. NRC. 2020k. "Locations of Low-Level Waste Disposal Facilities." Nuclear Regulatory Commission: Radioactive Waste, March 12, 2020. https://www.nrc.gov/waste/llw-disposal/licensing/locations.html.
U.S. NRC. 2020l. "Safeguard Categories of SNM." U.S. NRC. March 11, 2020. https://www.nrc.gov/security/domestic/mca/snm.html.
U.S. NRC. 2020m. "Backgrounder on Decommissioning Nuclear Power Plants." U.S. NRC. July 8, 2020. https://www.nrc.gov/reading-rm/doc-collections/fact-sheets/decommissioning.html.
U.S. NRC. 2021a. "Discontinuation of Rulemaking—Spent Fuel Reprocessing." SECY-21-0026. Policy Issue (Notation Vote). Washington, DC: U.S. Nuclear Regulatory Commission, March 5, 2021. https://www.nrc.gov/docs/ML2030/ML20301A388.pdf.
U.S. NRC. 2021b. "Spent Fuel Reprocessing." Federal Register, Notices, 86(143). July 29, 2021. https://www.govinfo.gov/content/pkg/FR-2021-07-29/pdf/2021-16173.pdf.
U.S. NRC. 2021c. "Defense in Depth." U.S. NRC. March 9, 2021. https://www.nrc.gov/reading-rm/basic-ref/glossary/defense-in-depth.html.
U.S. NRC. 2021d. "Locations of Power Reactor Sites Undergoing Decommissioning." U.S. Nuclear Regulatory Commission, November 10, 2021. https://www.nrc.gov/info-finder/decommissioning/power-reactor/index.html.
U.S. NRC. 2021e. "Fermi—Unit 1." U.S. Nuclear Regulatory Commission, March 9, 2021. https://www.nrc.gov/info-finder/decommissioning/power-reactor/enrico-fermi-atomic-power-plant-unit-1.html.
U.S. NRC. 2021f. "West Valley Demonstration Project." U.S. Nuclear Regulatory Commission, March 24, 2021. www.nrc.gov/info-finder/decommissioning/complex/wv.html.
U.S. NRC. 2021g. "Staff Requirements—SECY-19-0095—Discontinuation of Rulemaking—Enhanced Security of Special Nuclear Material," August 4, 2021. https://www.nrc.gov/docs/ML2121/ML21217A065.pdf.
U.S. NRC. 2021h. "Fuel Cycle—Physical Security Requirements for Facilities with Category II Quantities of Special Nuclear Material Information Sheet." U.S. Nuclear Regulatory Commission, October 8, 2021. https://www.nrc.gov/reactors/new-reactors/advanced/rulemaking-and-guidance/fuel-cycle.html.
U.S. NRC. 2022a. Fuel Qualification for Advanced Reactors—Final. NUREG-2246. Washington, DC: U.S. Nuclear Regulatory Commission. https://www.nrc.gov/docs/ML2206/ML22063A131.pdf.
U.S. NRC. 2022b. "Integral Molten Salt Reactor (IMSR)." U.S. Nuclear Regulatory Commission. May 31, 2022. https://www.nrc.gov/reactors/new-reactors/advanced/licensing-activities/pre-application-activities/imsr.html.
U.S. NRC. 2022c. "Backgrounder on Decommissioning Nuclear Power Plants." U.S. Nuclear Regulatory Commission. June 23, 2022. https://www.nrc.gov/reading-rm/doc-collections/fact-sheets/decommissioning.html.
U.S. NRC. 2022d. "Oklo Inc.—Denial of the Aurora Combined Operating License Application for Failure to Supply Information (EPID L-2020-NEW-0004 and EPID L-2020-NEW-0005)." Washington, DC: U.S. Nuclear Regulatory Commission. https://www.nrc.gov/docs/ML2135/ML21357A034.pdf.
U.S. NRC. n.d. "Nuclear Power Reactor Testing Needs and Prototype Plants for Advanced Reactor Designs." https://www.nrc.gov/docs/ML1702/ML17025A353.pdf. Accessed February 23, 2022.
U.S. Nuclear Waste Technical Review Board. 2017. "Vitrified High-Level Radioactive Waste." Revision 1, November 2017. https://www.nwtrb.gov/docs/default-source/facts-sheets/vitrified_hlw.pdf?sfvrsn=16.
Vaden, D. 2007. "Fuel Conditioning Facility Electrorefiner Model Predictions Versus Measurements." INL/CON-07-12980, Pre-print. Idaho Falls, ID: Idaho National Laboratory. October. https://inldigitallibrary.inl.gov/sites/sti/sti/3984768.pdf.
van den Akker, B. P., and J. Ahn. 2013. "Performance Assessment for Geological Disposal of Graphite Waste Containing TRISO Particles." Nuclear Technology 181(3):408-426. March 1. https://doi.org/10.13182/NT11-103.

Van der Stricht, S., and A. Janssens. 2010. Radioactive Effluents from Nuclear Power Stations and Nuclear Fuel Reprocessing Sites in the European Union, 2004-08. Radiation Protection 164. Luxembourg: European Commission. https://data.europa.eu/doi/10.2833/27366.

Van Rooyen, I. J., Subhashish M., and J. Rosales. 2018. "Effect of Neutron Irradiation Damage on Fission Product Transport in the SiC Layer of TRISO Fuel Particles." In Proceedings of HTR 2018. Warsaw, Poland. https://www.osti.gov/servlets/purl/1478815.

Vandegrift, G. F., M. C. Regalbuto, S. Aase, A. Bakel, T. J. Battisti, D. Bowers, J. P. Byrnes, M. A. Clark, D. G. Cummings, J. W. Emery, J. R. Falkenberg, A. V. Gelis, C. Pereira, L. Hafenrichter, Y. Tsai, K. J. Quigley, and M. H. Vander Pol. 2004 "Designing and Demonstration of the UREX+ Process Using Spent Nuclear Fuel." In ATALANTE 2004, 8. Nimes, France. https://inis.iaea.org/collection/NCLCollectionStore/_Public/36/008/36008582.pdf?r=1. Accessed April 21, 2022.

Vexler, A. 2021. "HALEU." Presentation to the National Academies' Committee on the Merits and Viability of Different Fuel Cycles and Technology Options and the Waste Aspects of Advanced Nuclear Reactors, online, September 15. https://www.nationalacademies.org/event/09-13-2021/merits-and-viability-of-different-nuclear-fuel-cycles-and-technology-options-and-the-waste-aspects-of-advanced-nuclear-reactors-meeting-9-september-13-15-2021.

Vine, G. 2011. "Abridged History of Reactor and Fuel Cycle Technologies Development: A White Paper for the Reactor and Fuel Cycle Technology Subcommittee of the Blue Ribbon Commission." Longenecker & Associates. https://cybercemetery.unt.edu/archive/brc/20120621004807/http://brc.gov/sites/default/files/documents/vine_abridged_history_of_reactor_and_fuel_cycle_technologies_development_final.pdf.

Von Baeckmann, A. 1988. "IAEA Safeguards in Nuclear-Weapon States." IAEA Bulletin 1. https://www.iaea.org/sites/default/files/publications/magazines/bulletin/bull30-1/30103552224.pdf.

von Hippel, F. 2007. "Managing Spent Fuel in the United States: The Illogic of Reprocessing." International Panel on Fissile Material. https://www.nrc.gov/docs/ML0817/ML081780779.pdf. Accessed April 21, 2022.

von Lensa, W., D. Vulpius, H.-J. Steinmetz, N. Girke, D. Bosbach, B. Thomauske, A.W. Banford, et al. 2011. "Treatment and Disposal of Irradiated Graphite and Other Carbonaceous Waste." ATW—Internationale Zeitschrift Fur Kernenergie 56(4-5):263-269.

von Lensa, W., G. Brinkmann, J. Lillington, and F. Shahrokhi. 2020. The status quo on HTGR decommissioning. Nuclear Engineering and Design 359: 110456. April 1. https://doi.org/10.1016/j.nucengdes.2019.110456.

Wade, D. C., and R. N. Hill. 1997. "The Design Rationale of the IFR." Progress in Nuclear Energy 31(1):13-42. January 1. https://doi.org/10.1016/0149-1970(96)00002-9.

Wallenius, J. 2021. "Fuel Cycle of SEALER-55." Presented at the Committee on Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Reactors, February 23. https://www.nationalacademies.org/event/02-22-2021/merits-and-viability-of-different-nuclear-fuel-cycles-and-technology-options-and-the-waste-aspects-of-advanced-nuclear-reactors-february-22-and-23-2021-meeting.

Wang, J., and S. Zhuang. 2019. "Extraction and Adsorption of U(VI) from Aqueous Solution Using Affinity Ligand-Based Technologies: An Overview." Reviews in Environmental Science and Bio/Technology 18(3):437-452. https://doi.org/10.1007/s11157-019-09507-y.

Westfall, C. 2004. "Vision and Reality: The EBR-II Story." Nuclear News (February):25-32. https://www.ne.anl.gov/About/reactors/EBR2-NN-2004-2-2.pdf. Accessed April 21, 2022.

Westinghouse Electric Company. 2003. "AP1000 Design Control Document, Rev. 3." APP-GW-GL-700. Westinghouse Electric Company.

Westphal, B. R., G. L. Fredrickson, G. G. Galbreth, D. Vaden, M. D. Elliott, J. C. Price, E. M. Honeyfield, M. N. Patterson, and L. A. Wurth. 2013. "Pyroprocessing of Fast Flux Test Facility Nuclear Fuel." In INL/CON-13-28161. Salt Lake City, UT. https://www.osti.gov/biblio/1111514.

White House. 1993. "Nonproliferation and Export Control Policy Statement." Fact sheet. September 27, 1993. Washington, DC: Executive Office of the President.

Wickham, A., H.-J. Steinmetz, P. O'Sullivan, and M. I. Ojovan. 2017. "Updating Irradiated Graphite Disposal: Project 'GRAPA' and the International Decommissioning Network." Journal of Environmental Radioactivity 171:34-40. May 1. https://doi.org/10.1016/j.jenvrad.2017.01.022.

Wigeland, R., T. Taiwo, M. Todosow, W. Halsey, and J. Gehin. 2009. "AFCI Options Study." INL/EXT-10-17639. Idaho Falls, ID: Idaho National Laboratory. September 1. https://doi.org/10.2172/978356. Accessed April 21, 2022.

Wigeland, R., T. Taiwo, H. Ludewig, M. Todosow, W. Halsey, J. Gehin, R. Jubin, et al. 2014. "Nuclear Fuel Cycle Evaluation and Screening—Final Report." Fuel Cycle Research & Development. U.S. Department of Energy, October 8, 2014. https://fuelcycleevaluation.inl.gov/Shared%20Documents/ES%20Main%20Report.pdf.

Williamson, M. A. 2020. "Metallic Fuel Recycling." Presentation to the National Academies' Committee on Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Reactors, December 8. https://www.nationalacademies.org/event/12-07-2020/merits-and-viability-of-different-nuclear-fuel-cycles-and-technology-options-and-the-waste-aspects-of-advanced-nuclear-december-7-8-2020-meeting.

Williamson, M., and T. Taiwo. 2021. Responses to the Committee's Additional Information Requests. September 13.

Willrich, M., and R. K. Lester. 1977. Radioactive Waste Management and Regulation. New York: Free Press.

WNA (World Nuclear Association). 2017a. "Mixed Oxide (MOX) Fuel." Updated October 2017. https://world-nuclear.org/information-library/nuclear-fuel-cycle/fuel-recycling/mixed-oxide-fuel-mox.aspx. Accessed April 21, 2022.

WNA. 2017b. "Lithium." World Nuclear Association. October 2017. https://world-nuclear.org/information-library/current-and-future-generation/lithium.aspx.

WNA. 2020a. "Uranium Enrichment." September. https://world-nuclear.org/information-library/nuclear-fuel-cycle/conversion-enrichment-and-fabrication/uranium-enrichment.aspx.

WNA. 2020b. "Generation IV Nuclear Reactors." December. https://www.world-nuclear.org/information-library/nuclear-fuel-cycle/nuclear-power-reactors/generation-iv-nuclear-reactors.aspx.

WNA. 2020c. "Thorium." November. http://www.world-nuclear.org/information-library/current-and-future-generation/thorium.aspx.

WNA. 2021a. "Uranium Mining Overview." September. https://world-nuclear.org/information-library/nuclear-fuel-cycle/mining-of-uranium/uranium-mining-overview.aspx.

WNA. 2021b. "U.S. Nuclear Fuel Cycle." May. https://world-nuclear.org/information-library/country-profiles/countries-t-z/usa-nuclear-fuel-cycle.aspx.

WNA. 2021c. "Nuclear Fuel and Its Fabrication." October. https://world-nuclear.org/information-library/nuclear-fuel-cycle/conversion-enrichment-and-fabrication/fuel-fabrication.aspx.

WNA. 2021d. "Fast Neutron Reactors." Updated August. https://world-nuclear.org/information-library/current-and-future-generation/fast-neutron-reactors.aspx.

WNA. 2021d. "Russia's Nuclear Fuel Cycle." Country Profile. Updated May. https://world-nuclear.org/information-library/country-profiles/countries-o-s/russia-nuclear-fuel-cycle.aspx.

WNA. 2021e. "Nuclear Power in the United Kingdom." Country Profile. Updated August. https://www.world-nuclear.org/information-library/country-profiles/countries-t-z/united-kingdom.aspx.

WNA. 2021f. "Nuclear Power in Japan." Country Profile. Updated June. https://world-nuclear.org/information-library/country-profiles/countries-g-n/japan-nuclear-power.aspx.

WNA. 2021g. "Nuclear Power in India." Country Profile. Updated June. https://world-nuclear.org/information-library/country-profiles/countries-g-n/india.aspx.

WNA. 2021h. "China's Nuclear Fuel Cycle." Country Profile. Updated March. https://world-nuclear.org/information-library/country-profiles/countries-a-f/china-nuclear-fuel-cycle.aspx.

WNA. 2021i. "Research Reactors." Updated June. https://world-nuclear.org/information-library/non-power-nuclear-applications/radioisotopes-research/research-reactors.aspx.

WNA. 2022a. "World Nuclear Association." https://world-nuclear.org. Accessed April 21, 2022.

WNA. 2022b. "Conversion and Deconversion." Updated January. https://world-nuclear.org/information-library/nuclear-fuel-cycle/conversion-enrichment-and-fabrication/conversion-and-deconversion.aspx.

WNA. 2022c. "Nuclear Power in France." Updated January. https://www.world-nuclear.org/information-library/country-profiles/countries-a-f/france.aspx.

WNA. 2022d. "Physics of Uranium and Nuclear Energy." February. https://world-nuclear.org/information-library/nuclear-fuel-cycle/introduction/physics-of-nuclear-energy.aspx.

WNISR (World Nuclear Industry Status Report). 2022. "World Nuclear Industry Status Report." https://www.worldnuclearreport.org.

WNN (World Nuclear News). 2010. "PBMR Facing Massive Cuts." February 18. https://www.world-nuclear-news.org/C_PBMR_facing_massive_cuts_1802101.html.

WNN. 2018. "Reprocessing Ceases at UK's Thorp Plant." World Nuclear News, November 14, 2018. https://www.world-nuclear-news.org/Articles/Reprocessing-ceases-at-UKs-Thorp-plant.

WNN. 2020a. "Congress Approves Nuclear Energy Funding for FY2021." World Nuclear News, December 23, 2020. https://world-nuclear-news.org/Articles/Congress-approves-nuclear-energy-funding-for-FY202.

WNN. 2020b. "Rosatom and US DOC Welcome Change to Uranium Agreement." World Nuclear News, October 8, 2020. https://world-nuclear-news.org/Articles/Rosatom-and-US-DOC-welcome-change-to-uranium-agree.

WNN. 2021a. "Joint Venture Formed to Spur SMR Deployment in Sweden." February 15, 2021. https://world-nuclear-news.org/Articles/Joint-venture-formed-to-spur-SMR-deployment-in-Swe

WNN. 2021b. "Chinese Fast Reactor Begins High-Power Operation." February 19, 2021. https://world-nuclear-news.org/Articles/Chinese-fast-reactor-begins-high-power-operation.

WNN. 2021c. "Preliminary Design of TRISO-X Fuel Plant Completed." World Nuclear News, November 2, 2021. https://www.world-nuclear-news.org/Articles/Preliminary-design-of-TRISO-X-fuel-plant-completed.

WNN. 2021d. "US Conversion Plant Gears Up for Next 40 Years." World Nuclear News, April 14, 2021. https://world-nuclear-news.org/Articles/US-conversion-plant-gears-up-for-next-40-years.

WNN. 2022a. "One-Year Delay in Canadian Repository Site Selection." World Nuclear News. August 15, 2022. https://www.world-nuclear-news.org/Articles/One-year-delay-in-Canadian-repository-site-selecti.

WNN. 2022b. "Chinese Molten-Salt Reactor Cleared for Start Up." World Nuclear News. August 9, 2022. https://www.world-nuclear-news.org/Articles/Chinese-molten-salt-reactor-cleared-for-start-up.

Wood, E. S., J. T. White, B. Jaques, D. Burkes, and P. Demkowicz. 2020. "10 - Advances in Fuel Fabrication." In Advances in Nuclear Fuel Chemistry, edited by M. H. A. Piro, pp. 371-418. Woodhead Publishing Series in Energy. Woodhead Publishing. https://doi.org/10.1016/B978-0-08-102571-0.00011-2.

Worrall, A. 2013. "Utilization of Used Nuclear Fuel in a Potential Future US Fuel Cycle Scenario." In WM2013 Conference. Phoenix, AZ. https://www.ourenergypolicy.org/wp-content/uploads/2013/09/Pub40177.pdf.

Worrall, A. 2021. "Rationale for Reprocessing and Recycling." Presentation to the National Academies' Committee on Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Reactors, June 2, 2021. https://www.nationalacademies.org/event/06-01-2021/merits-and-viability-of-different-nuclear-fuel-cycles-and-technology-options-and-the-waste-aspects-of-advanced-nuclear-reactors-meeting-6-part-1-june-1-2-2021.

Worth, R. N., A. Theodosiou, W. Bodel, J. D. Arregui-Mena, A. J. Wickham, A. N. Jones, and P. M. Mummery. 2021. "The Distribution and Selective Decontamination of Carbon-14 from Nuclear Graphite." Journal of Nuclear Materials 556:153167. December 1. https://doi.org/10.1016/j.jnucmat.2021.153167.

X-energy. 2022. "X-Energy's TRISO-X Selects Oak Ridge Horizon Center for First Commercial Advanced Reactor Fuel Fabrication Facility in North America." X-energy, April 4. https://x-energy.com/media/news-releases/x-energy-triso-x-selects-oak-ridge-horizon-center-for-first-commercial-advanced-reactor-fuel-fabrication-facility-in-north-america.

Xu, X., L. Xu, J. Ao, Y. Liang, C. Li, Y. Wang, C. Huang, et al. 2020a. "Ultrahigh and Economical Uranium Extraction from Seawater via Interconnected Open-Pore Architecture Poly(Amidoxime) Fiber." Journal of Materials Chemistry A 8(42):22032-22044. https://doi.org/10.1039/D0TA07180C.

Xu, S. G., S. Chester, K. Choi, and E. Bulemela. 2020b. "Characteristic Waste Streams from Small Modular Reactors Considered for Deployment in Canada." CNL Nuclear Review 9(1):83-92. https://pubs.cnl.ca/doi/full/10.1139/CNR.2019.00008.

Yang, W.S. 2012 "Fast Reactor Physics and Computational Methods." Nuclear Engineering and Technology 44(2):177-198. https://doi.org/10.5516/NET.01.2012.504.

Yardley, B. W. D., R. C Ewing, and R. A. Whittleston. 2016. "Deep-Mined Geological Disposal of Radioactive Waste." Elements: An International Magazine of Mineralogy, Geochemistry, and Petrology 12(4). August.

Zaetta, A., F. Varaine, Ch. Chabert, E. Dupont, J. F. Lebrat, and J. Tommasi. 2005. "La Faisabilité Scientifique de La Transmutation Des Déchets à Vie Longue." Commissariat à l'énergie atomique et aux énergies alternatives. https://inis.iaea.org/collection/NCLCollectionStore/_Public/38/041/38041742.pdf.

Zhan, L., Y. Bo, T. Lin, and Z. Fan. 2021. "Development and Outlook of Advanced Nuclear Energy Technology." Energy Strategy Reviews 34:100630. March 1. https://doi.org/10.1016/j.esr.2021.100630.

Zhang, H. 2021. "On China's Nuclear Fuel Cycle." Presentation to the National Academies' Committee on Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Reactors, December 6. https://www.nationalacademies.org/event/12-06-2021/merits-and-viability-of-different-nuclear-fuel-cycles-and-technology-options-and-the-waste-aspects-of-advanced-nuclear-reactors-meeting-12-december-6-7-2021.

Zhang, J., L. Li, S. He, and Y. Chen. 2016. "Calculation of Radioactivity and Dose Rate of Activated Corrosion Products in Water-Cooled Fusion Reactor." Science and Technology of Nuclear Installations 2016:6051834. https://doi.org/10.1155/2016/6051834.

Zhu, L., W. Duan, J. Xu, and Y. Zhu. 2012. "Uranium Extraction from TRISO-Coated Fuel Particles Using Supercritical CO_2 Containing Tri-n-Butyl Phosphate." Journal of Hazardous Materials 241-242:456-462. https://doi.org/10.1016/j.jhazmat.2012.09.072.

Zuidema, P. 2021. "Radioactive Waste Management Issues to Be Considered When Evaluating Different Nuclear Fuel Cycle and Reactor Variants." National Academies' Committee on Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Reactors, June 7. https://www.nationalacademies.org/event/06-07-2021/merits-and-viability-of-different-nuclear-fuel-cycles-and-technology-options-and-the-waste-aspects-of-advanced-nuclear-reactors-meeting-6-part-2-june-7-2021.

Appendix A

Biographical Sketches of the Committee and Project Staff

Janice Dunn Lee (*Chair*) retired in January 2017 after a 40-year career in nuclear policy and regulatory fields. She served for more than 30 years at the U.S. Nuclear Regulatory Commission in Washington, DC, in senior management positions focused on nuclear nonproliferation, national security, and nuclear safety policy. Her most senior position was director of international programs, a role in which she served for more than 8 years. This was followed by 10 years with international nuclear organizations in Paris, France, and Vienna, Austria. In particular, for almost 5 years, Ms. Dunn Lee served as deputy director general of the Nuclear Energy Agency of the Organisation for Economic Co-operation and Development. She then served a 5-year term as deputy director general and head of the department of management at the International Atomic Energy Agency and was the highest-ranking American in the United Nations (UN) agencies based in Vienna. In January 2019, Ms. Dunn Lee came out of retirement for a 1-year assignment to be the acting chief executive officer of the UN Joint Staff Pension Fund in New York City. She graduated from University of California, Berkeley, with a bachelor's degree in sociology and has a master's degree in international relations from the Fletcher School of Law and Diplomacy at Tufts University.

Patricia A. "Trish" Baisden (*Vice Chair*) retired in 2013 from the Lawrence Livermore National Laboratory (LLNL) after a career spanning more than 37 years. Since retiring, she has become an adjunct professor at San Jose State University and has been the primary instructor for the American Chemical Society/Department of Energy Summer School in Nuclear Chemistry, an intensive 6-week undergraduate fellowship program designed to introduce nuclear and radiochemical concepts to outstanding upper-level undergraduate science and engineering majors, and to stimulate their interest in pursuing graduate studies in the field. At the time of her retirement from LLNL, Dr. Baisden was deputy program manager for the Inertial Confinement Fusion Program, a position she assumed in 2012 upon the conclusion of the National Ignition Campaign (NIC). NIC was a multi-laboratory scientific and technology development program at LLNL focused on using the National Ignition Facility (NIF) to achieve ignition and thermonuclear burn in the laboratory via inertial confinement fusion. She also served as director for operations and LLNL institutional deputy for NIC. Dr. Baisden is a nuclear chemist, and during her career at LLNL, she held a number of technical and management positions, including division leader for analytical sciences, deputy director of the Seaborg Institute, materials program leader for NIF, chief scientist, and deputy associate director for the Chemistry and Material Sciences Directorate. She has served on numerous study panels and review committees, as an editor of the journal *Radiochimica Acta*, and as chair of the American Chemical Society's Division of Nuclear Chemistry and Technology. She has served on five National Academies' consensus

study committees. Dr. Baisden's research interests include nuclear fusion, lasers and optical materials, heavy ion reactions, heavy element fission properties, the chemistry of 4 and 5f elements, and nuclear power and advanced fuel cycles. She earned a B.S. in 1971 and a Ph.D. in 1975 in chemistry under Professor Gregory R. Choppin from Florida State University. Before joining the staff at LLNL, she held a 2-year postdoctoral appointment with Professor Glenn T. Seaborg at the University of California Lawrence Berkeley National Laboratory.

Margaret S. Y. Chu (NAE) provides consulting services to domestic and international clients in nuclear waste management, nuclear fuel cycle analysis, nuclear security analysis, and research and development. Her career has been devoted to promoting safe nuclear energy and nuclear fuel cycles. Dr. Chu's extensive experience includes successfully managing large, multidisciplinary projects and negotiating with customers, regulators, and stakeholders. For more than 20 years, she served at Sandia National Laboratories in several capacities, including as director of the Nuclear Waste Management Program Center, manager of the Environmental Risk Assessment and Waste Management Department, and deputy manager of the Waste Isolation Pilot Project (WIPP) and Technical Integration Department. In 2002, Dr. Chu was appointed by President George W. Bush as director of the U.S. Department of Energy's (DOE's) Office of Civilian Radioactive Waste Management, which is responsible for developing the nation's waste disposal system for spent nuclear fuel and high-level radioactive waste at Yucca Mountain. She has authored nearly 50 publications and has received numerous awards. Dr. Chu was a member of the National Academies' Nuclear and Radiation Studies Board, a member of the Advisory Committee of Reactor Safeguards at the U.S. Nuclear Regulatory Commission, and a member of the Nuclear Energy Advisory Committee at DOE. She was elected to the National Academy of Engineering in 2018 for her contributions for permanent disposal of radioactive waste in deep underground repositories. Dr. Chu holds a B.S. from Purdue University in chemistry and a Ph.D. from the University of Minnesota in physical (quantum) chemistry.

Paul T. Dickman is a senior policy fellow, based in Washington, DC, with Argonne National Laboratory, focusing on international nuclear energy, nonproliferation, and national security policy. He has held several senior leadership positions. At the U.S. Nuclear Regulatory Commission, Mr. Dickman served as chief of staff to Chairman Dale E. Klein. At the U.S. Department of Energy's National Nuclear Security Administration, he served on the undersecretary's staff as deputy director for the Office of Policy. Mr. Dickman is also an active member in the American Nuclear Society, currently chairs its External Affairs Committee, and served as study director for the Society's Special Committee Report on the nuclear accident at the Fukushima-Daiichi. He is also an advisor to the Japanese government agency responsible for the decommissioning of the Fukushima accident site and is president-elect for the World Council on Isotopes. Mr. Dickman serves on the National Academies' Nuclear and Radiation Studies Board and was a committee member for the National Academies' study Independent Assessment of Science and Technology for the Department of Energy's Defense Environmental Cleanup Program. He has published technical and policy papers on radioactive waste management, nuclear materials recycling and disposition, and international nuclear nonproliferation. Mr. Dickman holds a bachelor's degree in history (of science) and a master's degree of natural sciences in nuclear chemistry and physics.

Craig S. Hansen is an independent business consultant with 27 years of executive and senior-level experience in facility/site management; business and product line management; executing large and complex nuclear plant manufacturing, construction, decommissioning, and nuclear reactor servicing contracts; and successful leadership of complex technical projects facing a wide range of stakeholder challenges. He has extensive experience with BWXT, formerly the nuclear technology business of the Babcock & Wilcox Company (B&W). Mr. Hansen's most recent service was as president and board member (2013–2014) at B&W's American Centrifuge Manufacturing, LLC (ACM), where he was responsible for management and operations of the American Centrifuge Technology and Manufacturing Center located in Oak Ridge, Tennessee; there, his role included overseeing direction, management, and operation through bankruptcy and program realignment; managing a sophisticated technical manufacturing operation in a highly automated facility; and leading product-line diversification and demobilization due to government funding cuts. In B&W's nuclear manufacturing division (2008–2013), he was vice president of nuclear equipment, where he was responsible for B&W's global commercial nuclear equipment business along

with U.S. and Canadian manufacturing sites, worldwide contracts, and product lines. From 2003 through 2008, Mr. Hansen organized and managed B&W's government relations team. As B&W's deputy site manager (2001–2003), he accelerated the cleanup and public relations at the U.S. Department of Energy Miamisburg Environmental Management Project (Mound Plant), a site on the National Priorities List since 1989 because of past disposal practices and releases to the environment. Prior to B&W, he worked on the Naval Nuclear Propulsion Program in Washington, DC, and in Idaho (1988–2001) in a series of progressively responsible positions at the nuclear reactor headquarters and naval reactor site management. He also served as the first chairman of the U.S. Department of Commerce Civil Nuclear Trade Advisory Committee. Mr. Hansen has a B.A. from Eastern Washington University in operations management.

Edwin S. Lyman is director of the Nuclear Power Safety Project at the Union of Concerned Scientists (UCS) and is an internationally recognized expert on nuclear proliferation and nuclear terrorism, as well as nuclear power safety and security. He is a member of the Institute of Nuclear Materials Management and the American Nuclear Society, and has testified numerous times before Congress and the U.S. Nuclear Regulatory Commission. Since joining UCS in 2003, Dr. Lyman has published articles in a number of journals and magazines, including *Science*, the *Bulletin of the Atomic Scientists*, *Science and Global Security*, and *Arms Control Today*. He also co-authored the critically acclaimed book *Fukushima: The Story of a Nuclear Disaster* (New Press, 2014). In 2018, he was awarded the Leo Szilard Lectureship Award from the American Physical Society. Before joining UCS, Dr. Lyman was president of the Nuclear Control Institute, a Washington, DC–based organization focused on nuclear proliferation. From 1992 to 1995, he was a postdoctoral research associate at Princeton University's Center for Energy and Environmental Studies (now the Science and Global Security Program). Dr. Lyman earned a B.A. in physics from New York University in 1986 and a Ph.D. in physics from Cornell University in 1992.

Allison M. Macfarlane is currently professor and director, School of Public Policy and Global Affairs, Faculty of Arts, University of British Columbia (UBC). She has held both academic and government positions in the field of energy and environmental policy, especially nuclear policy. Most recently, Dr. Macfarlane directed the Institute for International Science and Technology Policy at The George Washington University. She recently held a fellowship at the Wilson International Center for Scholars in Washington, DC, and was Fulbright distinguished chair in applied public policy at Flinders University and Carnegie Mellon Adelaide in Australia. Dr. Macfarlane was the first geologist (and the third woman) to chair the U.S. Nuclear Regulatory Commission, a position she held from 2012 to 2014. She has held fellowships at Radcliffe College, the Massachusetts Institute of Technology (MIT), Stanford University, and Harvard University, and she has been on the faculty at Georgia Tech in earth science and international affairs, at George Mason University in environmental science and policy, and in the Elliott School of International Affairs at The George Washington University. From 2010 to 2012, Dr. Macfarlane served on the White House Blue Ribbon Commission on America's Nuclear Future, created by the Obama administration to recommend a new national policy on high-level nuclear waste. She has also served on National Academies' committees on nuclear energy and nuclear weapons issues, and she has chaired the Science and Security Board of the *Bulletin of the Atomic Scientists*, the group that sets the publication's famous "doomsday clock." In 2006, MIT Press published a book she co-edited, *Uncertainty Underground: Yucca Mountain and the Nation's High-Level Nuclear Waste*. Dr. Macfarlane has published extensively in *Science*, *Nature*, *Environmental Science and Technology*, the *Bulletin of the Atomic Scientists*, and other journals. Dr. Macfarlane's research has focused on technical, social, and policy aspects of nuclear energy production and nuclear waste management and disposal, as well as regulation, nuclear nonproliferation, and energy policy. She holds a B.S. from the University of Rochester and a Ph.D. in Earth science from MIT.

Albert J. Machiels retired from the Electric Power Research Institute (EPRI) in June 2017. During his 35-year tenure at EPRI, his responsibilities included the oversight and/or management of several research and development programs in the following technical areas: Nuclear Fuel Industry Research (NFIR), Severe Accident Issue Resolution, Severe Accident Technology, Control and Diagnostics, Advanced Light Water Reactor, Primary System Corrosion Research, Pressurized Water Reactor Materials Reliability Program, Boiling Water Reactor

Vessel Integrity Program, Risk and Reliability, Balance-of-Plant Corrosion, and Used Fuel and High-Level Waste Management. From 1996 to 2017, Dr. Machiels was actively engaged in topics related to spent fuel management (storage and transportation) and advanced fuel cycles, areas in which his contributions gained international recognition. He served as EPRI executive liaison to the Nuclear Management and Resources Council (NUMARC, now part of the Nuclear Energy Institute) from August 1988 to October 1989. Before joining EPRI, Dr. Machiels was a tenured, associate professor of nuclear engineering at the University of Illinois at Urbana-Champaign. Prior to coming to the United States in 1970, he was a lecturer at the University of Liège, Belgium. Dr. Machiels has authored more than 200 publications and technical contributions, and has managed and contributed to the publication of more than 100 EPRI reports. He has served on several national and international panels and committees. In April 2012, Dr. Machiels received a Lifetime Achievement Award in recognition of his strategic and technical contributions to the nuclear generation of electricity. He was only the eighth individual (and the first one from the Nuclear Generation Sector) to receive such an award in the 40-plus-year history of EPRI. Presently, he is a part-time consultant on topics related to the management of spent nuclear fuel. Dr. Machiels received Ingénieur Civil Chimiste and Ingénieur en Génie Nucléaire degrees from the University of Liège, in Belgium, and a Ph.D. in engineering from the University of California, Berkeley.

Christophe Poinssot is nuclear counselor at the French Embassy in China and the Commissariat à l'Énergie Atomique et aux Énergies Alternatives (CEA) representative at the embassy, where he has served since 2018. In November 2020, he was appointed deputy general director and scientific director of the French Geological Survey (BRGM, Bureau de Recherche Géologiques et Minières). Dr. Poinssot has more than 25 years of experience in the CEA, where he occupied various positions. As a CEA representative in China, he was in charge of developing partnerships in industry and research and development in the field of low-carbon, numeric and digital, and health technologies. From 2011 to 2018, Dr. Poinssot was head of the Research Department on Mining and Fuel Recycling Processes at the Nuclear Energy Division, CEA Marcoule. Prior to joining this department in 2008, he spent 15 years in CEA Saclay working in the field of nuclear waste and used fuel management. In particular, he launched and coordinated the French research program on spent nuclear fuel long-term evolution and headed the CEA labs working on radionuclides migration in the environment. Since 2011, he has been a professor of nuclear chemistry at the National Institute of Nuclear Science and Technology (INSTN) and has been appointed CEA international expert in fuel cycle and actinides chemistry. Dr. Poinssot has long been involved in international collaborations and has been a member of several scientific committees of international conferences and for the evaluation of research teams. In particular, he was a steering committee member of the study "Reset of America's Nuclear Waste Management Strategy and Policy," which published its report in 2018. In 2017, Dr. Poinssot was awarded Officer from the French order "Palmes Académiques" and also received the Van Geen Prize and Chair from SCK-CEN (Belgium). He has authored more than 60 international articles and 110 presentations in international scientific conferences. His most recent research includes developing a holistic approach to assess the relative sustainability of the different fuel cycles and nuclear systems. He earned two master's degrees in 1995 at the École Normale Supérieure de Paris in Earth sciences and material sciences, and a Ph.D. in 1997 in material sciences at the Université Pierre and Marie Curie (Paris).

Jeffrey D. Semancik has served as director of the Radiation Division for the Connecticut Department of Energy and Environmental Protection since joining the state in July 2014. In this role, he is responsible for policy development, decision making, program implementation, compliance and enforcement in the areas of radiation safety, radiological emergency preparedness and response, radioactive materials licensing and inspection, nuclear and radioactive waste management, and transportation of nuclear and radioactive material. Mr. Semancik serves as a member of the State of Connecticut's Nuclear Energy Advisory Council, as compact secretary for the New England Radiological Health Committee, as state liaison officer to the U.S. Nuclear Regulatory Commission, as alternate commissioner for the State of Connecticut on the Atlantic Interstate Low-Level Radioactive Waste Compact, as representative to the Northeast High Level Radioactive Waste Transportation Task Force at the Council of State Governments, and as a Nuclear Advisory Committee member for Three Rivers Community College (Norwich, Connecticut). He serves on the board (and was chairperson in 2019) of the Conference of Radiation Control Program Directors, and

is a member of the Government Coordinating Council to the Cyber and Infrastructure Security Agency's (CISA's) Nuclear Sector Critical Infrastructure Protection Advisory Council. He is also an advisor on the Committee for Evaluation of Guidelines, Resources, and Tools for Radiological and Nuclear Emergency Response and Recovery and chairperson of the Committee on Naval Nuclear Propulsion. Mr. Semancik is qualified as and has contributed to the development of the Radiological Operations Support Specialist for emergency response to radiological emergencies, including participating in various multiagency nuclear power plant, improvised nuclear device, and radiological dispersal device exercises. Prior to state service, he worked in commercial nuclear power for 23 years in a variety of positions, including licensed senior reactor operator, operations manager, engineering director, and plant manager. Mr. Semancik began his career as an officer in the U.S. Navy serving on board the nuclear-powered aircraft carrier *USS Abraham Lincoln* and the guided missile destroyer *USS Scott* during Operation Desert Shield. He holds a B.S. in physics from the U.S. Naval Academy, an M.S. in electrical engineering from Rensselaer Polytechnic Institute, and an M.B.A. from the University of Connecticut.

Ken B. Sorenson retired from Sandia National Laboratories in January 2018 after 35 years of service. For 23 years, he served as technical manager, overseeing technology development for the back end of the commercial nuclear fuel cycle. The focus of this work was on long-term storage and transportation of spent nuclear fuel, aspects of which included materials degradation of fuel cladding, response of storage and transportation packages to severe mechanical and thermal environments, safeguards and security of nuclear materials, systems analyses, and regulatory analyses. Mr. Sorenson led the development and managed the national laboratory component of the U.S. Department of Energy's Office of Nuclear Energy (DOE-NE) program to assess the integrity of spent nuclear fuel under long-term storage and transportation conditions. This program, in collaboration with the DOE national laboratories, the U.S. Nuclear Regulatory Commission (U.S. NRC), and industry, is still active today. Mr. Sorenson managed the work at Sandia for the U.S. NRC after the 9/11 attack to assess the radiological consequence of various potential terrorist attacks on licensed storage and transportation systems. From 2013 to 2014, he served as president of the Institute of Nuclear Materials Management, an international organization focused on instilling best practices in the areas of nuclear materials safeguards, security, and safety management around the world. Mr. Sorenson was technical editor and wrote two chapters of a reference book titled *Safe and Secure Transport and Storage of Radioactive Materials* and wrote and coauthored numerous journal articles and conference presentations. Currently, he is consulting with Sandia in the areas of knowledge transfer, technical report writing and reviewing, and strategic planning support. Mr. Sorenson earned a B.S. in 1975 in civil engineering from the University of Arizona, an M.S. in 1982 in civil (structural) engineering from Colorado State University, and an M.B.A. in 2000 from the University of New Mexico. He is a licensed professional engineer in the state of Colorado (P.E. 0018367).

Nathalie A. Wall is an internationally recognized radiochemist. Her research focuses on the environmental behavior of radionuclides for applications to the nuclear fuel cycle and nuclear forensics. She is best known for her work on the thermodynamic properties of actinides and fission products. Dr. Wall worked in the Department of Nuclear Waste Management at the Commissariat à l'Énergie Atomique (CEA) and as staff scientist at Sandia National Laboratories. She was a faculty member of the Chemistry Department at Washington State University prior to moving to the University of Florida's Department of Materials Science and Engineering in 2019. A native of France, Dr. Wall earned her undergraduate degree in physical sciences in 1989 at the University of Paris. She completed her Ph.D. with Professor R. Guillaumont and Dr. V. Moulin in radiochemistry at the University of Paris in 1993 and was a postdoctoral research associate with Professor G. R. Choppin at Florida State University.

PROJECT STAFF

Charles D. Ferguson (*Study Director*) is senior board director of the Board on Chemical Sciences and Technology and the Nuclear and Radiation Studies Board in the Division on Earth and Life Studies at the National Academies of Sciences, Engineering, and Medicine. Previously, he was president of the Federation of American Scientists (FAS). Prior to FAS, Dr. Ferguson was Philip D. Reed senior fellow for science and technology at the Council on

Foreign Relations (CFR), where he specialized in nuclear issues and served as project director for the Independent Task Force on U.S. Nuclear Weapons Policy chaired by William J. Perry and Brent Scowcroft. Before CFR, he was scientist-in-residence at the Monterey Institute's Center for Nonproliferation Studies, where he coauthored the book *The Four Faces of Nuclear Terrorism* (Routledge, 2005) and was lead author of the January 2003 report *Commercial Radioactive Sources: Surveying the Security Risks*. For his work on security of radioactive sources, Dr. Ferguson was awarded the Robert S. Landauer Memorial Lecture Award from the Health Physics Society in 2003. He is also the author of *Nuclear Energy: What Everyone Needs to Know* (Oxford University Press, 2011). In addition, he has worked as a physical scientist in the Office of Nuclear Safety at the U.S. Department of State, and served as a nuclear engineering officer and submarine officer in the U.S. Navy. Dr. Ferguson is an elected fellow of the American Physical Society in recognition of his service to public policy and public education on nuclear issues. He earned a B.S. in physics with distinction from the U.S. Naval Academy, and an M.A. and a Ph.D., both also in physics, from Boston University.

Ourania "Rania" Kosti is a senior program officer at the Nuclear and Radiation Studies Board (NRSB) of the National Academies of Sciences, Engineering, and Medicine. Her interests within the NRSB focus on radiation health effects, and she is principal investigator for the National Academies' Radiation Effects Research Foundation Program, which supports studies of the atomic bombing survivors in Japan. Prior to her current appointment, Dr. Kosti was a postdoctoral fellow at the Lombardi Comprehensive Cancer Center at Georgetown University Hospital in Washington, DC, where she conducted research on biomarker development for early cancer detection using case-control epidemiological study designs. She focused primarily on prostate, breast, and liver cancers, trying to identify those individuals who are at high risk of developing malignancies. Dr. Kosti also trained at the National Cancer Institute (2005–2007). She received a B.S. in biochemistry from the University of Surrey, United Kingdom; an M.S. in molecular medicine from the University College London; and a Ph.D. in molecular endocrinology from St. Bartholomew's Hospital in London, United Kingdom.

Catherine F. Wise (*Co-Study Director from July 2022*) is a program officer with the Board on Energy and Environmental Systems at the National Academies of Sciences, Engineering, and Medicine. She contributes to projects related to advanced nuclear fuel cycles and reactors, vehicle fuel economy technologies, decarbonization of the energy system, and carbon dioxide utilization technologies and infrastructure. Prior to the National Academies, Dr. Wise worked as a graduate research assistant studying proton-coupled electron transfer in electrochemical systems. She received a B.S. in chemistry from the College of William and Mary in 2015 and a Ph.D. in chemistry from Yale University in 2020.

Appendix B

Presentations at the Committee's Information-Gathering Meetings

PUBLIC MEETING #1: SEPTEMBER 21–22, 2020, VIRTUAL

Presentations from the U.S. Department of Energy's Office of Nuclear Energy

- *Opening Remarks on Behalf of the Department of Energy's Office of Nuclear Energy (DOE-NE),* William Boyle, Acting Deputy Assistant Secretary, Office of Spent Fuel and Waste Disposition, U.S. Department of Energy
- *Perspectives on the Congressional Mandate,* Christopher T. Hanson, Commissioner, U.S. Nuclear Regulatory Commission and Former Staff Member, U.S. Senate Appropriations Committee
- *U.S. House of Representatives' Appropriations Staff Presentation on the Congressional Mandate,* Scott McKee, Professional Staff Member, U.S. House Appropriations Committee, Subcommittee on Energy and Water
- *Overview of Nuclear Fuel Cycle and Supply Chain Program,* Andrew Griffith, Deputy Assistant Secretary, Office of Nuclear Fuel Cycle and Supply Chain, U.S. Department of Energy
- *Support for Advanced Reactor Development and Deployment,* Alice Caponiti, Deputy Assistant Secretary, Office of Reactor Fleet and Advanced Reactor Deployment, U.S. Department of Energy
- *Overview of Nuclear Fuel Cycle Evaluation and Screening,* Bhupinder P. Singh, Program Manager, Nuclear Fuel Cycle and Supply Chain (NE-4), Office of Nuclear Energy, U.S. Department of Energy
- *Overview of the U.S. Nuclear Regulatory Commission's Regulatory Programs on Fuel Cycles and Waste Aspects of Advanced Reactors,* Christopher M. Regan, Deputy Director, James Hammelman, Senior Chemical Process Engineer, and Jose R. Cuadrado, Project Manager, Division of Fuel Management, Office of Nuclear Material Safety and Safeguards

PUBLIC MEETING #2: DECEMBER 7–8, 2020, VIRTUAL

Presentations from Argonne National Laboratory and Idaho National Laboratory

- *Versatile Test Reactor (VTR) Project Status and Overview,* Thomas O'Connor, Director, Versatile Test Reactor Program, Office of Nuclear Energy, U.S. Department of Energy

- *VTR Core Design and Fuel Selection,* Thomas Fanning, Manager, Safety and Engineering Analysis Department, Nuclear Science and Engineering Division, Argonne National Laboratory
- *VTR Experimental Capabilities,* Kevan Weaver, Director, Experimental Capabilities for the VTR, Idaho National Laboratory
- *VTR Fuel Cycle and Waste Management,* Douglas Crawford, Director of the Transient Reactor Test Facility (TREAT), Idaho National Laboratory
- *Systems Perspective on Advanced Fuel Cycles and Waste Management,* Temitope Taiwo, Interim Director, Nuclear Science and Engineering Division, Argonne National Laboratory
- *Summary of DOE-NE Material Recovery and Waste Form Campaign R&D,* Terry Todd, Laboratory Fellow, Idaho National Laboratory
- *Ceramic Fuel and TRISO Fuel Recycle Options,* Terry Todd, Laboratory Fellow, Idaho National Laboratory
- *Molten Salt Reactor Fuel Recycle Options,* Candido Pereira, Deputy Director, Chemical and Fuel Cycle Technologies Division, Argonne National Laboratory
- *Metal Fuel Recycle Options,* Mark Williamson, Division Director, Chemical and Fuel Cycle Technologies Division, Argonne National Laboratory
- *Waste Form Development,* Bill Ebert, Manager, Pyroprocess and Waste Form Development, Chemical and Fuel Cycle Technologies Division, Argonne National Laboratory
- *Nonproliferation Considerations,* Michael Miller, Director, Nuclear Nonproliferation, Idaho National Laboratory
- *Economics of Future Fuel Cycle Options,* Brent Dixon, Deputy National Technical Director, DOE-NE Systems Analysis and Integration, R&D Campaign, Idaho National Laboratory

PUBLIC MEETING #3: JANUARY 11 AND 13, 2021, VIRTUAL

- *NuScale Power: A Scalable Clean Energy Solution,* José Reyes, Chief Technology Officer and Cofounder, NuScale Power
- *Terrestrial Energy: Overview of the Integral Molten Salt Reactor,* David LeBlanc, President, Chief Technology Officer and Director, Terrestrial Energy
- *ThorCon 1 GW Plant,* Lars Jorgensen, Chief Executive Officer, ThorCon
- *BWXT's Advanced Nuclear Reactor,* Erik Nygaard, Director of Research and Engineering, BWXT Advanced Technologies
- *Overview of X-Energy's 200 MWth Xe-100 Reactor,* Eben Mulder, Chief Nuclear Officer, X-Energy
- *General Atomics,* Christina Back, Vice President, Nuclear Technologies and Materials, and Robert Schleicher, Chief Engineer, General Atomics
- *Kairos Power Overview,* Edward Blandford, Chief Technology Officer and Cofounder, and Per Peterson, Chief Nuclear Officer and Cofounder, Kairos
- *Flibe Energy,* Kirk Sorensen, President and Chief Technologist, Flibe Energy

PUBLIC MEETING #4: FEBRUARY 22–23, 2021, VIRTUAL

- *Framatome Steam Cycle High-Temperature Gas-Cooled Reactors,* Lewis Lommers, HTGR Engineering Manager, Framatome
- *Natrium,* Pavel Hejzlar, TerraPower Technical Fellow, and Tara Neider, SVP Program Development and Lab Facilities, TerraPower
- *TerraPower's Molten Chloride Fast Reactor (MCFR),* Jeff Latkowski, Senior Vice President (Innovation) and Program Director of Molden Chloride Fast Reactor Program, TerraPower
- *The ARC-100 Advanced SMR,* Ed Arthur, Vice President, Fuel Cycle Management and Safeguards, and John Sackett, Senior Vice President and Chief Technology Officer, ARC Clean Energy
- *LeadCold,* Janne Wallenius, Project Leader, SEALER Blykalla/LeadCold
- *Moltex Technology,* Rory O'Sullivan, CEO for North America, Moltex Energy

- *Oklo,* Caroline Cochran, COO and Co-Founder, and Jacob DeWitte, CEO and Cofounder, Oklo
- *X-Energy's TRISO Fuel Production,* Pete Pappano, Vice President, Fuel Production, X-Energy

PUBLIC MEETING #5: MAY 17–19, 2021, VIRTUAL

- *Application of Safeguards by Design to Advanced Reactors,* Jeremy Whitlock, Section Head for Concepts and Approaches, Department of Safeguards, International Atomic Energy Agency
- *Perspectives on Proliferation-Resistance (and Terror-Resistance): Fuel Cycles and Advanced Reactors,* Matthew Bunn, Co–Principal Investigator, Managing the Atom Project, Belfar Center, Harvard University
- *Safeguards and Security Analysis for Fuel Cycle Facilities,* Ben Cipiti, Sandia National Laboratories
- *New Approaches Utilizing Process Monitoring Data and Machine Learning,* Ben Cipiti, Sandia National Laboratories
- *Advanced Reactor Safeguards and Security,* Ben Cipiti, Sandia National Laboratories
- *IAEA Safeguards Considerations Associated with HALEU,* Warren Stern, Deputy Chair, Nonproliferation and Security, Brookhaven National Laboratory
- *Aspects of Material Accounting and Control for Advanced Reactors,* Robert K. Larsen, Senior Nuclear Security Officer, Nuclear Security of Materials and Facilities Section, Division of Nuclear Security, Department of Nuclear Safety and Security, International Atomic Energy Agency
- *Proliferation Risks of Laser Enrichment of Uranium,* Ryan Snyder, Visiting Fellow, University of Hamburg, Institute of Peace Research and Security Policy
- *Defense Nuclear Nonproliferation and Cooperation with the Advanced Reactor Industry,* Jeffery Chamberlin, Associate Assistant Deputy Administrator, Office of Material Management and Minimization, Office of Defense Nuclear Nonproliferation, National Nuclear Security Administration, U.S. Department of Energy
- *Defense Nuclear Nonproliferation's Civil Nuclear Security Program,* Katherine C. Holt, Program Director for Analytics and Innovation, Office of International Nuclear Security, National Nuclear Security Administration, U.S. Department of Energy
- *Support to U.S. Advanced Reactor Stakeholders,* Anagha Iyengar, Program Manager, Office of International Nuclear Safeguards, National Nuclear Security Administration, U.S. Department of Energy
- *HALEU Security Theft and Diversion at Fixed Sites and in Transit,* Tim Harris, Senior Program Manager, Office of Nuclear Security and Incident Response (NSIR), U.S. Nuclear Regulatory Commission
- *Safeguards and Material Accounting for Advanced Reactors and Associated Fuel Cycles,* James Rubenstone, Chief of the Material Control and Accounting Branch in the Division of Fuel Safety, Safeguards, and Environmental Review, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission

PUBLIC MEETING #6: JUNE 2 AND JUNE 7, 2021, VIRTUAL

- *Status and Prospects of HALEU Production in the United States,* Daniel B. Poneman, President and CEO, and Larry Cutlip, Senior Vice President, Field Operations, Centrus Energy Corporation
- *The Rationale for Reprocessing and Recycling,* Andrew Worrall, Section Head, Integrated Fuel Cycle, Oak Ridge National Laboratory, Deputy Director, Gateway for Accelerated Innovation in Nuclear, and UK Country Coordinator, DOE Office of Nuclear Energy
- *Impacts of Nuclear Fuel Cycle Choices on Permanent Disposal of High-Activity Radioactive Waste,* Peter N. Swift, Sandia National Laboratories
- *Radioactive Waste Management Issues to Be Considered When Evaluating Different Nuclear Fuel Cycle Options,* Piet Zuidema, Chief Scientific Officer, European Joint Programme on Radioactive Waste Management (EURAD)
- *Some Impacts of Advanced Fuel Cycle Options on Waste Management and Long-Term Disposal Risks,* Bernd Grambow, Professor of Excellence and Chair of Nuclear Waste Management, IMT Atlantique, École des Mines Télécom Atlantique, Nantes, France

- *Economic and Environmental Costs and Benefits of Reprocessing,* Frank N. von Hippel, Senior Research Physicist and Professor of Public and International Affairs, Emeritus, Princeton University
- *The Economics of Reprocessing and Recycling Versus Direct Disposal of Spent Nuclear Fuel,* Matthew Bunn, Co–Principal Investigator, Managing the Atom Project, Belfer Center, Harvard University
- *Westinghouse eVinci™ Micro-Reactor,* Vefa N. Kucukboyaci

PUBLIC MEETING #7: JULY 20, 2021, VIRTUAL

- *Attractiveness of Materials in Advanced Nuclear Fuel Cycles,* Charles G. Bathke, Los Alamos National Laboratory

PUBLIC MEETING #8: SEPTEMBER 13–15, 2021, VIRTUAL

- *Advanced Research Project Agency—Energy (ARPA-E) Efforts Supporting Advanced Nuclear,* Jenifer Shafer, Program Director, ARPA-E
- *Proliferation Resistance Using Methodology of the International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO),* Brian Boyer, Section Head, International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) Section, Division of Nuclear Power, Department of Nuclear Energy, International Atomic Energy Agency (IAEA)
- *Implementation of IAEA Safeguards within the United States,* David H. Hanks, Senior International Nuclear Safeguards Analyst, U.S. Nuclear Regulatory Commission
- *Front End of the Nuclear Fuel Cycle,* Amir Vexler, President and CEO, Orano USA
- *Back End of the Nuclear Fuel Cycle,* Arnaud Gay, Executive VP, Technical Department and International Operations, Orano, and Sven Bader, Technical Consultant, Orano Federal Services
- *Russia's Efforts to Develop Advanced Nuclear Fuel Cycles,* Pavel Podvig, Independent Analyst, Geneva, Switzerland

PUBLIC MEETING #9: SEPTEMBER 28–29, 2021, VIRTUAL

- *Overview of the Generation IV International Forum,* Hideki Kamide, Deputy Director General for Sector of Fast Reactor and Advanced Reactor Research and Development, Japan Atomic Energy Agency (JAEA), and Chair of the Generation IV International
- *Status of the SFR Technology Developments in the Generation IV International Forum: Reactor Design and Fuel Research and Development,* Gilles Rodriguez, Technical Director of the Generation IV International Forum, and Frédéric Serre, Chair of the Generation IV International Forum Sodium-Cooled Fast Reactor (SFR) System Steering Committee
- *Review of Spent Fuel Reprocessing and Associated Accident Phenomena,* Fred Gelbard, Chemical Engineer, Sandia National Laboratories
- *Meeting Advanced Reactors' Needs for High-Assay Low-Enriched Uranium*: (1) U.S. Department of Energy's Plans and (2) Independent Report with Results from Gateway for Accelerated Innovation in Nuclear April 2020 Workshop; (1) Andrew Griffith, Deputy Assistant Secretary, Office of Nuclear Fuel Cycle and Supply Chain, U.S. Department of Energy, and (2) Monica Regalbuto, Director, Nuclear Fuel Cycle Strategy, Idaho National Laboratory
- *Role of Aqueous Separations in Advanced Nuclear Fuel Cycles,* Gregg Lumetta, Lab Fellow, Chemist, Nuclear Chemistry and Engineering Group, Pacific Northwest National Laboratory
- *France's Strategy on the Back-End of the Fuel Cycle and the Management and Disposal of Radioactive Waste,* Patrick Landais, High Commissioner for Atomic Energy, CEA, French Alternative Energies and Atomic Energy Commission
- *Treatment of EBR-II Spent Fuel,* Michael N. Patterson, Program Manager, Used Fuels Treatment, Materials and Fuels Complex, Idaho National Laboratory

- *Panel Discussion with Utility Executives,* Greg Cullen, Vice President for Energy Services and Development, Energy Northwest; Nick Irvin, Research and Development Director, Advanced Energy Systems, Cross Cutting Technology, and Strategy, Southern Company; Marilyn C. Kray, Vice President of Nuclear Strategy and Development, Exelon; and Chris Nolan, Vice President of Regulatory Affairs, Policy, and Emergency Preparedness, Duke Energy

PUBLIC MEETING #10: OCTOBER 14–15, 2021, VIRTUAL

- *Canada's Nuclear Waste Management Organization (NWMO) Perspectives on Waste Acceptance from Advanced Reactors,* Paul Gierszewski, NWMO Director, Safety and Technical Research, Canada
- *The U.S. Government Accountability Office's Reports on Uranium Management,* Allison Bawden, Director in the U.S. Government Accountability Office's Natural Resources and Environment Team
- *Sodium-Cooled Fast Reactor Technologies,* Robert Hill, Program Manager, Advanced NE R&D and Argonne Distinguished Fellow, Argonne National Laboratory
- *Spent Nuclear Fuel Management and Back-End Safeguards,* Rowen Price, Research Assistant, Nuclear Safeguards, and Cindy Vestergaard, Senior Fellow and Director, Block Chain in Practice and Nuclear Safeguards, Stimson Center
- *Safeguards Technology Considerations and Research Needs for Thorium Fuel Cycles and Molten Salt Reactors,* Louise G. Worrall, Senior R&D Scientist, Non-Destructive Measurement Science and Technology Group, Nuclear Nonproliferation Division, National Nuclear Security Sciences Directorate, Oak Ridge National Laboratory
- *Nuclear Waste from Small Modular Reactors,* Lindsay Krall, Geochemist, Swedish Nuclear Fuel and Waste Management Company
- *Hazards Associated with Molten Salt Reactor Systems,* Joanna Mcfarlane, Chemist, Nuclear Energy and Fuel Cycle Division, Oak Ridge National Laboratory
- *Westinghouse's Lead-Cooled Fast Reactor,* Paolo Ferroni, Fellow Engineer and Technical Lead, Westinghouse Lead Fast Reactor Project, and Fausto Fraceschini, Consulting Engineer, Westinghouse Mangiarotti, Westinghouse Electric Company

PUBLIC MEETING #11: DECEMBER 6, 2021, VIRTUAL

- *China's Nuclear Fuel Cycle Programs,* Hui Zhang, Senior Research Associate, Project on Managing the Atom, Belfer Center for Science and International Affairs, Harvard Kenny School
- *Trends in Nuclear Fuel Cycle Workforce,* Terry A. Todd, Emeritus Fellow, Idaho National Laboratory
- *Perspectives from the Department of Energy's Office of Nuclear Energy,* Kathryn D. Huff, Acting Assistant Secretary and Principal Deputy Assistant Secretary for the Office of Nuclear Energy, U.S. Department of Energy
- *Perspectives from Congress*: Panel Discussion with Key Staff of the Appropriations Committees of the Senate and House of Representatives—Aaron Goldner, Professional Staff Member, Senate Appropriations Committee, and Scott McKee, Professional Staff Member, House Appropriations Committee

PUBLIC MEETING #12: DECEMBER 16, 2021, VIRTUAL

- *Advanced Non–Light Water Reactors: Integrated Waste Management System Considerations,* Mark Nutt, Deputy National Technical Director of the DOE-NE Integrated Waste Management System

Appendix C

Acronyms and Abbreviations

ADS	accelerator-driven system
AEC	Atomic Energy Commission
AFC	Advanced Fuels Campaign
AFCI	Advanced Fuel Cycle Initiative
AGR	advanced gas reactor
AIP	American Institute of Physics
AIROX	Atomics International Reduction Oxidation
ALWR	advanced light water reactor
ANDRA	National Radioactive Waste Management Agency (*L'Agence Nationale pour la gestion des Déchets RadioActifs*) (France)
ANL	Argonne National Laboratory
ANS	American Nuclear Society
ANSI	American National Standards Institute
ARC	Advanced Reactor Concept
ARDP	Advanced Reactor Demonstration Program
ARPA-E	Advanced Research Projects Agency–Energy
ARS	advanced reactor safeguards
ASTRID	Advanced Sodium Technological Reactor for Industrial Demonstration
ATF	accident-tolerant fuel
ATR	Advanced Test Reactor
AVR	*Arbeitsgemeinschaft Versuchsreaktor* (Joint Working Group Experimental Reactor) (Germany)
AVR-TLK	dry storage canister (*Tritium Laboratory Karlsruhe*) (Germany)
B205	Alternative name for Magnox Reprocessing Plant at Sellafield (United Kingdom)
BANR	BWXT Advanced Nuclear Reactor
BCG	Boston Consulting Group
BISO	bistructural isotropic particle fuel
BN	fast sodium (reactor) (БН - Быстрый Натриевый) (in Russian abbreviations)
BNFL	British Nuclear Fuels Limited

BNL	Brookhaven National Laboratory
BRC	2012 Blue Ribbon Commission on America's Nuclear Future
BREST-OD	Lead-cooled fast reactor—Experimental Demonstration (БРЕСТ-ОД–Быстрый Реактор Естественной безопасности со Свинцовым Теплоносителем—Опытно-Демонстрационный (in Russian abbreviations)
BWR	boiling water reactor
BWXT	Originally Babcock and Wilcox Technologies
C/S	containment and surveillance
CANDU	Canadian Deuterium Uranium
CARBOWASTE	European Commission's Carbonaceous Waste program
CB&I	CB&I Areva MOX Services, LLC, in 2002, now MOX Services, LLC, formerly known as Chicago Bridge and Ironworks
CBO	U.S. Congressional Budget Office
CEA	Alternative Energies and Atomic Energy Commission (*Commissariat à l'Énergie Atomique et aux Énergies Alternatives*) (France)
CEFR	China Experimental Fast Reactor
CFR	United States Code of Federal Regulations
CIGEO	Industrial Centre for Geological Disposal (*Centre Industriel de Stockage GEOlogique*) (France)
CNSC	Canadian Nuclear Safety Commission
COEX™	Co-extraction of uranium and plutonium
CORAIL	fuel assembly design contains both LEU and MOX rods (*COmbustible Recyclage A ILot*) (France)
CPF	chemical process facility
CPP	chemical processing plant
CPPNM	1979 Convention on the Physical Protection of Nuclear Material (international treaty) and the 2005 Amendment
CR	conversion ratio
CRBR	Clinch River Breeder Reactor
CRS	Congressional Research Service
CSA	comprehensive safeguard agreement
CTR	CarboThermic Reduction
CTR-N	CarboThermic Reduction of uranium oxide followed by Nitridation (synthesis)
D&D	decontamination and decommissioning
DA	destructive analysis
DBT	design basis threat
DECON	decontamination (and dismantling) (decommissioning strategy)
DIAMEX	diamide extraction
DNFSB	Defense Nuclear Facilities Safety Board
DNN	Office of Defense Nuclear Nonproliferation (within the U.S. Department of Energy's National Nuclear Security Administration)
DOC	U.S. Department of Commerce
DOD	U.S. Department of Defense
DOE	U.S. Department of Energy
DOE-NE	U.S. Department of Energy's Office of Nuclear Energy
DOS	U.S. Department of State
dpa	displacements per atom
EBR	experimental breeder reactor
EC	European Commission

EDF	Électricité de France, electric utility company
EFL	Eligible Facilities List
EG	evaluation group (of fuel cycle)
Eh	reduction-oxidation (redox) potential versus the standard hydrogen electrode
EIA	U.S. Energy Information Administration
EIS	Environmental Impact Statement
EMT	electrometallurgical treatment
ENTOMB	radioactive elements are entombed on site, such as with concrete (decommissioning strategy)
EPA	U.S. Environmental Protection Agency
EPACT	Energy Policy Act of 2005
EPR	European Pressurized Water Reactor
EPRI	Electric Power Research Institute
ESA	European Supply Agency
EU	enriched uranium
eV	electron volt
EXAm	EXtraction (recovery) of Americium (process)
FBR	fast breeder reactor
FBR-MOX	fast breeder reactor-mixed oxide
FCCI	fuel-cladding chemical interaction
FFH	fusion-fission hybrids
FFTF	Fast Flux Test Facility
FHR	fluoride-cooled high-temperature reactor
fima	fissions per initial metal atom
FLiBe	lithium fluoride (LiF) and beryllium fluoride (BeF), includes $2LiF\text{-}BeF_2$ and $^7Li_2BeF_4$
FONSI	Finding of No Significant Impact
FP	fission product
FR	fast reactor
FR-MOX	fast reactor-mixed oxide (fuel)
FSV	Fort St. Vrain (reactor site)
FY	fiscal year
GA	General Atomics
GAIN	Gateway for Accelerated Innovation in Nuclear
GANEX	Grouped ActiNide EXtraction (process)
GAO	U.S. Government Accountability Office
GE	General Electric
GEH	General Electric Hitachi
GFR	gas-cooled fast reactor
GIF	Generation-IV International Forum
GMODS	Glass Material Oxidation and Dissolution System
GNEP	Global Nuclear Energy Partnership
GRAPA	GRAphite Processing Approaches
GSG	IAEA's General Safety Guide
GTCC	Greater than Class C
GWd	gigawatt-day
GWe	gigawatt electrical
Gy	gray
H-D-N	hydride-dehydride-nitride (synthesis)

HAC	hypothetical accident conditions
HALEU	high-assay low-enriched uranium
HDND	high-dose neutron detector
HEPA	high efficiency particulate air (filter)
HEU	highly enriched uranium
HI-STORE	Holtec International-Storage (Consolidated Interim Storage Facility)
HLRW	high-level radioactive waste
HLW	high-level (radioactive) waste
HM	heavy metal
HT	high tensile (steel coding)
HTGR	high-temperature gas-cooled reactor
HTR	high-temperature (gas-cooled) reactor
HTR-PM	high-temperature (gas-cooled) reactor—pebble-bed modular
I-NERI	International Nuclear Energy Research Initiative
IAEA	International Atomic Energy Agency
IFR	Integral Fast Reactor
ILW	intermediate-level waste
IMMONET	IAEA repository for data and reports for irradiated graphite (knowledge network)
IMSR	integral molten salt reactor
INEPC	Office of International Nuclear Energy Policy and Cooperation (within the U.S. Department of Energy's Office of Nuclear Energy)
INFCIRC	International Atomic Energy Agency's Information Circular (news communication)
INL	Idaho National Laboratory
INS	Office of International Nuclear Safeguards (within the U.S. Department of Energy's National Nuclear Security Administration)
IPFM	International Panel on Fissile Materials
iPWR	integral pressurized water reactor
IPyC	internal pyrolytic carbon
ISFSI	Independent Spent Fuel Storage Installation
J	joule
JAEA	Japan Atomic Energy Agency
JCO	Japan Consulting Office, formerly Japan Nuclear Fuel Conversion Co.
JNFL	Japan Nuclear Fuel Limited
KAERI	Korea Atomic Energy Research Institute
KBS	nuclear fuel safety (*kärnbränslesäkerhet*) (Sweden)
k_{eff}	effective neutron multiplication factor
kgHM	kilogram heavy metal
KP-FHR	Kairos Power Fluoride Salt-Cooled High Temperature Reactor
KTH	Kungliga Tekniska högskolan (Royal Institute of Technology) (Sweden)
ky	kilo years (1,000 years)
LANL	Los Alamos National Laboratory
LCC	liquid cadmium cathode
LCOE	levelized cost of electricity
LEU	low-enriched uranium
LFR	lead-cooled fast reactor
LFTR	liquid fluoride thorium reactor

LLEA	local law enforcement agencies
LLFP	long-lived fission products
LLNL	Lawrence Livermore National Laboratory
LLRW	low-level radioactive waste
LLW	low-level (radioactive) waste
LWR	light water reactor
LWT	legal weight truck (cask)
M	mega (as in MCi = megacurie)
m^3	cubic meters
MA	minor actinides
MC&A	material control and accounting
MCFR	molten chloride fast reactor
MCi	megacurie
MCRE	Molten Chloride Reactor Experiment
MeV	mega electron volt
MFFF	MOX Fuel Fabrication Facility
Mg	megagram
MIGHTR	modular integrated gas-cooled high temperature reactor
MIT	Massachusetts Institute of Technology
MOX	mixed oxide
MOXEUS	Mixed OXide Enriched Uranium Support, also known as MIX (MIxed oXide [enriched uranium support])
MPACT	Materials Protection, Accounting, and Control Technologies (U.S. Department of Energy campaign)
MRIE	maximum routine inspection effort
MSR	molten salt reactor
MSR-Cl	molten salt reactor—chloride
MSR-F	molten salt reactor—fluoride
MT	metric ton
MTHM	metric tons heavy metal
MTU	metric ton of uranium
MTW	Honeywell Metropolis Works (conversion plant)
MUF	material unaccounted for
MWd	megawatt-day
MWe	megawatts electric
MWh	megawatt hour
MWth	megawatts thermal
n-stamp	nuclear qualified firm
NAC	NAC International, Inc. (formerly Nuclear Assurance Corporation)
NAE	National Academy of Engineering
NAGRA	National Cooperative for the Disposal of Radioactive Waste (*NAtionale Genossenschaft fuer die lagerung Radioaktiver Abfaella*) (Switzerland)
NASAP	U.S. Department of Energy's Nonproliferation Alternative Systems Assessment Program
NDA	nondestructive assay
NEA-OECD	Nuclear Energy Agency of the Organisation for Economic Co-operation and Development
NEAC	Nuclear Energy Advisory Committee
NED	nuclear explosive device
NEI	Nuclear Engineering International

NEICA	Nuclear Energy Innovation Capabilities Act of 2017
NEIMA	Nuclear Energy Innovation and Modernization Act of 2019
NEM	nuclear-explosive material
NEXT	New EXtraction system for TRU Recovery (process)
NFCE&S	nuclear fuel cycle evaluation and screening
NFWG	Nuclear Fuel Working Group
NGSAM (red)	Next Generation Systems Analysis Model
NHI	Nuclear Hydrogen Initiative
NNL	National Nuclear Laboratory
NNSA	National Nuclear Security Administration
NPM	NuScale Power Module
NPR	nuclear power reactor
NPT	1968 Treaty on the Non-Proliferation of Nuclear Weapons
NUREG	Nuclear Regulatory Report (U.S. Nuclear Regulatory Commission)
NWPA	Nuclear Waste Policy Act of 1982
NWTRB	Nuclear Waste Technical Review Board
OECD	Organisation for Economic Co-operation and Development
ONWARDS	Optimizing Nuclear Waste and Advanced Reactor Disposal Systems (U.S. Department of Energy's Advanced Research Project Agency-Energy program)
OPyC	outer pyrolytic carbon
ORNL	Oak Ridge National Laboratory
OTA	Office of Technology Assessment
OTC	once-through cycle
P&T	partitioning and transmutation
PBq	petabecquerel
PBR	pebble-bed reactor
pH	scale of acidity or alkalinity (base) (*potenz hydrogen*, respectively)
PM	pebble-bed module
PPE	personal protective equipment
PRISM	Power Reactor Innovative Small Module
PUREX	plutonium uranium reduction extraction
PWR	pressurized water reactor
PyC	pyrolytic carbon
Q-brine	quinary salt mine brine
R&D	research and development
rad	radiation absorbed dose
RE	reactive elements (Fe-10Cr-4Al-RE)
rem	equivalent dose (Roentgen Equivalent Man [or human]) (predominantly United States)
REMIX fuel	REgenerated MIXture of U-Pu oxides
RIS	regulatory issue summaries (U.S. Nuclear Regulatory Commission)
RRP	Rokkasho Reprocessing Plant (Japan)
RT	radiotoxicity
S&T	separation and transmutation
SAFSTOR	SAFe STORage deferred dismantling (decommissioning strategy)
SANEX	Selective ActiNide Extraction

SC-HGTR	Framatome's Steam-Cooled High-Temperature Gas-Cooled Reactor
SCKCEN	StudieCentrum voor Kernenergie·Centre d'Etudes Nucléaire (Belgium)
SCWR	supercritical water-cooled reactor
SDO	standards developing organization
SEALER	SwEdish Advanced LEad Reactor
SECY	commission paper (SECretarY of the U.S. Nuclear Regulatory Commission)
SFR	sodium-cooled fast reactor
SISUS	Subgroup on IAEA Safeguards in the United States
SMR	small modular reactor
SNF/HLW	spent nuclear fuel and high-level radioactive waste
SNL	Sandia National Laboratories
SNM	special nuclear materials
SQ	significant quantities (of special nuclear materials)
SRS	Savannah River Site
SSAC	State System of Accounting for and Control (of nuclear material)
SSNM	strategic special nuclear materials
SSR-W	stable salt reactor—wasteburner
STATS report	acronym for separations technology and transmutation systems, the 1996 National Research Council report, *Nuclear Wastes: Technologies for Separations and Transmutation*
SUBATECH	Subatomic Physics and Associated Technologies (*Laboratoire de Physique SUBAtomique et des TECHnologies Associées*) (France)
Sv	Sieverts
SVBR	Lead-Bismuth Fast Reactor (SVBR—Свинцово-Висмутовый Быстрый Реактор) (in Russian abbreviations)
SWU	separative work unit
$t_{1/2}$	half-life (of a radioactive substance)
TALSPEAK	Trivalent Actinide Lanthanide Separation with Phosphorus-reagent Extraction from Aqueous Komplexes
TBP	tributyl phosphate
TBq	terabecquerel
TECDOC	International Atomic Energy Agency's TEChnical DOCument
TENEX	uranium products supplier (Techsnabexport (Техснабэкспорт) (Russia)
THORP	THermal Oxide Reprocessing Plant
TRISO	TRistructural ISOtropic
TRL	technical readiness level
TRU	transuranic elements
TRUEX	TRansUranium Extraction (process)
U_3O_8	triuranium octoxide
UC	uranium carbide
UCO	uranium oxycarbide
U_{dep}	depleted uranium
UF_6	uranium hexafluoride
UK	United Kingdom
UN	United Nations
UNF	used nuclear fuel
UO_2	uranium dioxide
UOX	uranium oxide
U_{rep}	reprocessed uranium

UREX	uranium recovery by extraction
U.S. NRC	U.S. Nuclear Regulatory Commission
U.S.C.	United States Code of Law
VHTR	very high-temperature reactor
VOA	voluntary offer agreement
VTR	versatile test reactor
W/THM	watts per ton of heavy metal
WIPP	Waste Isolation Pilot Plant
WNA	World Nuclear Association
WNN	World Nuclear News
WTP	Waste Treatment Plant
WVDP	West Valley Demonstration Project

Appendix D

Radioactive Waste Classifications and Waste Characteristics from Different Stages of the Fuel Cycle in the United States

In the United States, categories of radioactive waste include high-level waste, spent nuclear fuel, transuranic waste, low-level waste (which is further separated into Class A, Class B, Class C, and Greater Than Class C), depleted uranium, and uranium mill tailings. These classifications, defined in Table D.1, are generally based on the origin of the waste, but also on isotopic composition, radiotoxicity level, and concentration, and dictate how the waste must be treated, stored, transported, and disposed. Note that the United States does not have an Intermediate Level Waste (ILW) category as is the practice internationally[1]; the closest U.S. category to ILW is Greater-Than-Class C (GTCC) waste, as defined in Table D.1.

WASTES FROM DIFFERENT STAGES OF THE FUEL CYCLE

Radioactive waste streams are generated throughout the fuel cycle, regardless of fuel and reactor type, including from uranium mining and milling, uranium enrichment, fresh fuel fabrication, reactor operation, reprocessing operations (if applicable), recycle fuel fabrication (if applicable), and decommissioning. The relative amounts (volumes) of waste generated from each stage of the nuclear fuel cycle differ significantly, as shown in Table D.2. This section provides a brief overview of the types of wastes generated during each stage of the nuclear fuel cycle; more detailed discussions of new and unique wastes that may be generated from advanced reactors and fuel cycles can be found in Chapter 5.

Wastes from the Front End of the Fuel Cycle

Wastes from the front end of the fuel cycle include those generated from mining and milling, enrichment, and fuel fabrication. In the United States, uranium mining and milling wastes are classified as Naturally Occurring Radioactive Material (NORM) by the U.S. Nuclear Regulatory Commission, not as radioactive waste; therefore,

[1] According to the IAEA General Safety Guide, No. GSG-1, "Classification of Radioactive Waste": "Intermediate Level Waste is waste that, because of its content, particularly of long lived radionuclides, requires a greater degree of containment and isolation than that provided by near surface disposal. However, ILW needs no provision, or only limited provision, for heat dissipation during its storage and disposal. ILW may contain long lived radionuclides, in particular, alpha emitting radionuclides that will not decay to a level of activity concentration acceptable for near surface disposal during the time for which institutional controls can be relied upon. Therefore, waste in this class requires disposal at greater depths, of the order of tens of meters to a few hundred meters."

TABLE D.1 Radioactive Waste Categories and Definitions Used in the United States

Category	Definition	Source
High-Level Waste	"(A) the highly radioactive material resulting from the reprocessing of spent nuclear fuel, including liquid waste produced directly in reprocessing and any solid material derived from such liquid waste that contains fission products in sufficient concentrations; and (B) other highly radioactive material that the Commission, consistent with existing law, determines by rule requires permanent isolation."	Nuclear Waste Policy Act of 1982, 42 U.S.C. 10101
Spent Nuclear Fuel	"fuel that has been withdrawn from a nuclear reactor following irradiation, the constituent elements of which have not been separated by reprocessing"	Nuclear Waste Policy Act of 1982, 42 U.S.C. 10101
Transuranic Waste	"waste containing more than 100 nanocuries of alpha-emitting transuranic isotopes, with half-lives greater than twenty years, per gram of waste, except for: (1) high-level radioactive wastes; (2) wastes that the Department has determined, with the concurrence of the Administrator, do not need the degree of isolation required by this part; or (3) wastes that the Commission has approved for disposal on a case-by-case basis in accordance with 10 CFR Part 61."	40 CFR 191.02
Low-Level Waste	"waste that is not high-level radioactive waste, spent nuclear fuel, transuranic waste, byproduct material (as defined in section 11e.(2) of the Atomic Energy Act of 1954, as amended), or naturally occurring radioactive material."	Radioactive Waste Management Manual, DOE M 435.1-1
Class A	Most radioactivity from relatively short-lived radionuclides that reach background levels in several decades. See 10 CFR 61.55(a)(2)(i) and 10 CFR 61.56(a).	DOE-EM, 2019
Class B	Contains mostly short-lived radionuclides but at higher concentrations than Class A. See 10 CFR 61.56.	DOE-EM, 2019
Class C	Contains both short- and long-lived radionuclides, and at higher concentrations than Class B. See 10 CFR 61.56.	DOE-EM, 2019
Greater-Than-Class C	Contains higher concentrations of short- and long-lived radionuclides than Class C and sometimes requires disposal in geologic repository. See 10 CFR 61.55(a)(2)(iv) and 10 CFR 61.58.	DOE-EM, 2019
Depleted Uranium	"the source material uranium in which the isotope uranium-235 is less than 0.711 weight percent of the total uranium present."	10 CFR 40.4
Uranium Mill Tailings	"the remaining portion of a metal-bearing ore after some or all of such metal, such as uranium, has been extracted."	40 CFR 192.01

TABLE D.2 Relative Characteristics of Fuel Cycle Generated Waste

Waste-Generating Process	Relative Volume	Relative Activity	Relative Radiotoxicity
Enrichment	Small	Low	Low
Fuel fabrication	Small	Low	Low
Plant operation	Large	Medium	Low
Reprocessing	Small/Medium	High/Very high	High
Spent nuclear fuel for disposal	Medium	High	High
Decommissioning	Very large	Low	Very low

SOURCE: NEA-OECD (2006b).

they are not included in this report. Enrichment plants are characterized by low specific activity of the materials treated; they generate only low-level waste (LLW) and only in relatively small amounts. A modern centrifuge plant generates about 4×10^{-5} m^3/SWU of LLW (NEA-OECD, 2006b), where SWU is separative work unit, or the amount of separation done by an enrichment process. The amount of waste that will be generated by the production of high-assay low-enriched uranium (HALEU) is unknown at this time. Most of the U.S. fuel fabrication experience is in uranium oxide fuels, a process that generally involves the preparation and formation of fuel pellets, followed by the

assembly of those pellets into pins. LLW "from fuel fabrication includes filter media from water cleanup, waste oils, spent acids and bases, spent analytical solutions," cleaning solutions, and discarded scrap metals and equipment. The volume of LLW from fuel fabrication is estimated to be 0.6 m^3/metric tons of heavy metal (IAEA, 2019a).

Wastes from Reactor Operations

The nature and amount of waste produced during nuclear power plant operation "depend on the type of reactor, its specific design features, its operating conditions," maintenance activities performed, and the fuel integrity. The radioactive waste consists of activated structures and components, "moderator and coolant materials, corrosion products captured" in plant components, filters and resins, maintenance materials and consumables, and fission product contamination arising from activated corrosion product transport or leaks in the fuel. "Corrosion products, which originate within the reactor core or out of the reactor and are carried through the reactor core by the coolant or moderator, are neutron activated isotopes" (NEA-OECD, 2006b). Within 8 days of shut down, these activated isotopes decay by about an order of magnitude and are almost all short lived (Zhang et al., 2016). "Both fission products and activation products are distributed throughout the coolant and moderator systems, becoming the primary contributors to liquid processing waste and decontamination and maintenance waste" (NEA-OECD, 2006b). When nonwater coolants such as liquid sodium are used, new unique waste streams are generated because of the hazardous characteristics of liquid sodium.

Wastes from Decommissioning of Power Plants

Decommissioning nuclear power plants generates a significant amount of LLW, which results from the removal of fuel and coolant, the dismantling of the reactor structure, the containment and associated buildings, components, and equipment. This LLW includes mostly

- "activated equipment and materials, such as reactor internals, reactor vessel, and concrete shielding surrounding the reactor vessel";
- "corrosion products, such as deposits formed from corrosion and release into circulating coolants" and deposits on the various coolant system surfaces;
- environmental media contaminated from accumulation of radioactive isotopes released during operations, including soil, water, and building materials;
- on-site waste filtration and removal systems, such as off-gas charcoal adsorption beds (NEA-OECD, 2006b).

Most of the decommissioning wastes from advanced reactors may be similar to that of conventional reactors; however, waste from innovative coolant and reactor cores may differ significantly, as discussed in Chapter 5.

WASTE TREATMENT

Upon its initial generation, waste is usually treated directly at the facility to reduce volume for storage at the facility, reduce operational releases to the environment, and/or facilitate its further conditioning for storage or disposal. Waste treatments are well established and commercially available for most common waste streams, and vast operational experience has been accumulated.

Solid waste treatment technologies are used to predominantly reduce the waste volume to facilitate storage or reduce disposal costs. Aqueous liquid waste is verified to be within operational limits prior to discharge and diluted via the release pathway. Some aqueous liquid waste treatment technologies achieve further reductions in the bulk volume of liquid waste or in the amount of radioactivity discharge, mostly through evaporation or by separation of radionuclides using various filtration and sorption techniques (with the exception of tritium). The latter yields a substantially smaller volume of filtration materials and absorbents retained on site, although it results in higher concentrations of radionuclides. Gaseous waste streams are monitored to verify that releases are within limits and can be held for short-term decay prior to release or captured with filters and charcoal absorbers.

Treated waste also often requires additional conditioning before final disposal, with a goal of providing proper isolation of radioactive materials from the environment. Conditioning processes include transforming the waste into more stable solid forms or incorporating it into an inert matrix. For liquid waste conditioning, cementation and bituminization are well-matured and commonly used technologies.

WASTE DISPOSAL REQUIREMENTS AND MANAGEMENT PROGRAMS

Different disposal requirements exist for the different waste categories described in Table D.1. Transuranic waste generated from the nuclear defense program has been disposed of at the Waste Isolation Pilot Plant (WIPP) since 1999 (DOE, n.d.-b). Low-level activity waste streams are being or will be disposed of in near-surface disposal sites at four facilities: EnergySolutions Barnwell Operations in Barnwell, South Carolina; U.S. Ecology in Richland, Washington; EnergySolutions Clive Operations in Clive, Utah; and Waste Control Specialists, LLC, in Andrews, Texas (U.S. NRC, 2020k). DOE's inventory of depleted uranium hexafluoride, located primarily at the Paducah Site in Kentucky and the Portsmouth Site in Ohio, is in the process of being converted to depleted uranium oxide, which is more stable and therefore better suited for disposal (U.S. DOE, 2016). Uranium mine and mill tailings also require specific disposal sites to prevent long-term contamination of soil, air, and groundwater around inactive mining and milling sites; for example, the Moab Uranium Mill Tailings Remedial Action Project is transporting approximately 15.5 million tons of tailings from the Moab site to a disposal cell near Crescent Junction, Utah (DOE-EM, 2021).

Per the Nuclear Waste Policy Act of 1982 (NWPA) (Public Law 97-425), high-level waste and spent nuclear fuel must ultimately be disposed of in a geologic repository, but they are currently stored at 113 sites across the country (Peters et al., 2020) since no such repository exists. Table D.3, reproduced from Peters et al. (2020), depicts the projected growth of spent nuclear fuel that will have to be managed and disposed of for the scenarios of (1) assuming 60 years of operations of the fleet of light water reactors as of 2012, (2) addition of two new reactor builds, (3) shutdown at end of current license period of 40 years, and (4) addition of two new builds plus 8 additional current reactors obtaining a second 20-year license renewal to 80 years of operations. For any of these scenarios, the projected inventory will exceed the current legislated capacity of Yucca Mountain's repository, if licensed. Thus, an additional repository will be required.

TABLE D.3 Projected U.S. Commercial Spent Nuclear Fuel Inventories Through 2082 Under Different Scenarios[a]

	Fuel Discharges as of 12/31/2012		Forecast Discharges 1/1/2013 to 12/31/2019		Forecast Future Discharges 1/1/2020 to 12/31/2082		Total Projected Discharged Fuel		Delta from Reference	
Scenario	Assy.	Initial Uranium (MT)	Assy.	Initial Uranium (MT)	Assy.	Est. Initial Uranium (MT)	Assy.	Initial Uranium (MT)	Assy.	Initial Uranium (MT)
Reference Scenario 60-year operation unless announced otherwise	240,138	69,187	52,954	15,154	174,811	51,539	467,903	135,880	N/A	N/A
Scenario 1 Addition of 2 new builds	240,138	69,187	52,954	15,154	181,983	54,573	475,075	139,914	**7,172**	**3,034**
Scenario 2 Shutdown at end of current license period	240,138	69,187	52,954	15,154	164,369	48,431	457,461	132,772	**(10,442)**	**(3,108)**
Scenario 3 Addition of 2 new builds and 8 additional reactors obtaining a second 20-year license renewal	240,138	69,187	52,954	15,154	192,587	57,928	485,679	142,269	**17,776**	**6,389**

[a] Includes commercial light water reactor inventory and Morris and U.S. Department of Energy sites, other than Three Mile Island-Unit 2 fuel debris.
SOURCE: Reproduced from Table 2-22 of Peters et al. (2020). Courtesy of Savannah River National Laboratory.

Appendix E

Fuel Cycle Characteristics and Geologic Repository Metrics of Advanced Nuclear Reactors

KEY FEATURES OF NUCLEAR FUEL CYCLE

The fuel composition and neutron flux spectrum play a major role in determining fuel cycle characteristics of various advanced reactor designs reviewed by the committee and discussed in Chapters 3 and 4. The neutron flux spectrum, in turn, depends heavily on the fuel composition and coolant used as a heat transfer medium in the reactor core of the nuclear power plant. The flux spectrum represents the energy distribution of neutrons resulting from various neutron/nucleus collision processes that reduce the energy of neutrons produced from the fission process by ~2.0 MeV. The total energy produced in a fission process involving the splitting of a nucleus is approximately 200 MeV, with 2.5 neutrons released on average carrying an average energy of 2.0 MeV each.

In thermal-spectrum reactors (i.e., pressurized water reactors [PWRs] and boiling water reactors [BWRs] and cores, cooled by normal water, technically referred to as light water in contrast to heavy water), fission neutrons undergo collisions with hydrogen nuclei (i.e., protons), resulting in efficient slowing down or moderation of neutrons to arrive at an average neutron energy of 0.25 eV. Thus, water serves both as a coolant and moderator in light water reactors (LWRs). In contrast, in fast-spectrum reactors (e.g., sodium-cooled systems), neutrons have fewer chances of losing energy through collisions with the heavier sodium nuclei, thereby resulting in an average neutron energy of 0.1-0.2 MeV. The typical neutron flux spectrum $\phi(E)$ as a function of neutron energy E is compared for thermal and fast-spectrum reactors in Figure E.1, where the flux spectrum is represented as $\phi(u) = E\phi(E)$ in terms of neutron lethargy u, a variable equivalent to energy E.

The probability of fission reaction increases for a nucleus with an even number of protons and an odd number of neutrons such that even with a negligible kinetic energy of incident neutrons, the resulting compound nucleus may undergo a fission process and release neutrons and energy. The primary examples of nuclides with the particular combination of proton and neutron numbers are ^{233}U, ^{235}U, ^{239}Pu, and ^{241}Pu, known as *fissile nuclides*. On the other hand, a nucleus with an even number of both protons and neutrons usually requires neutrons with kinetic energy $E \geq 0.1$ MeV to undergo fission. Nuclides of this type are referred to as *fertile nuclides* (e.g., ^{234}U, ^{238}U, ^{240}Pu, and ^{242}Pu). A fertile nucleus upon neutron capture typically undergoes a radiative capture process resulting in a fissile nucleus. The fission process typically results in two lighter nuclei known as *fission products* (FPs) (Lee, 2020).

With approximately 200 MeV of energy released per fission process, the inventory of fission products in used nuclear fuel (UNF) is determined essentially by the thermal energy generated in the fuel discharged.

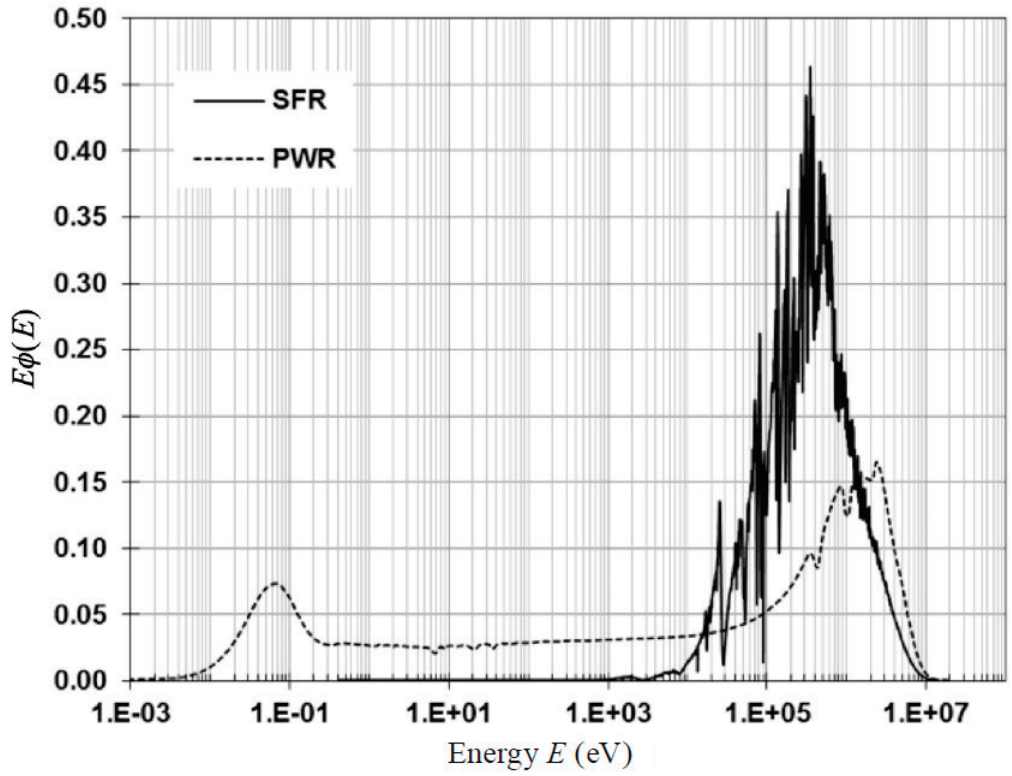


FIGURE E.1 Comparison of neutron flux spectrum for fast and thermal reactors.
NOTE: PWR = pressurized water reactor; SFR = sodium-cooled fast reactor.
SOURCE: Lee (2020).

Neutron reaction cross sections representing neutron/nucleus reaction probabilities—in particular, fission cross section and radiative capture cross section—depend heavily on the energy of neutrons and hence the neutron flux spectrum, as well as on the type of nuclides with which the neutrons interact. Table E.1 compares key spectrum-average neutron cross sections in units of barn [b] for typical thermal and fast-spectrum reactors. Note that the fission cross section is generally much larger in the thermal spectrum, while the ratio ^{239}Pu fission cross section to ^{238}U capture cross section is 8.1 in the fast spectrum compared with 34–98 for the thermal spectrum. This indicates that a fast-spectrum reactor requires a large fissile enrichment of either ^{239}Pu or ^{235}U to attain a critical system. This feature, at the same time, also allows enhanced neutron captures in fertile nuclides (e.g., converting fertile ^{238}U to fissile ^{239}Pu), serving as the basis for a fast-spectrum breeder reactor.

TABLE E.1 Spectrum-Average Cross Sections for Thermal and Fast-Spectrum Reactors

	Thermal Spectrum		Fast Spectrum
Reaction	PWR	VHTR	SFR
^{238}U capture σ_{γ}^{238} (b)	0.9	4.8	0.20
^{239}Pu fission σ_{f}^{239} (b)	89.2	165.0	1.65
$\sigma_{f}^{239}/\sigma_{\gamma}^{238}$	97.7	34.3	8.14

NOTE: PWR = pressurized water reactor; SFR = sodium-cooled fast reactor; VHTR = very-high-temperature reactor.
SOURCE: Yang (2012).

GROUPING ADVANCED REACTORS FOR FUEL CYCLE CHARACTERISTICS

Representative reactor concepts are analyzed with a focus on material flow sheets in an effort to present and compare fuel cycle characteristics of various advanced reactor designs under development. The material flow sheets are constructed primarily with the data presented in the 2014 *Nuclear Fuel Cycle Evaluation and Screening* (*NFCE&S*) report (Wigeland et al., 2014). The flow sheets reflect three once-through cycles contrasted with a design requiring continuous fuel processing and a two-tier system combining thermal and fast-spectrum reactors. The analysis covers primarily three major concepts that have been supported by the U.S. Department of Energy following the Generation IV Roadmap release.

Fuel cycle characteristics are represented primarily in terms of the fuel composition and coolant that play a dominant role in determining the neutron flux spectrum in the reactor core. The flux spectrum, together with the fuel composition, in turn characterizes the depletion and isotopic evolution of nuclear fuel materials. This observation suggests classifying advanced reactor designs as (a) fast-spectrum reactor, (b) gas-cooled reactor, and (c) molten salt reactor, together with (d) modular LWR, as summarized in Table E.2.

TABLE E.2 Key Attributes of Advanced Nuclear Reactors Under Development

Project Description	Power	Features	Fuel Cycle
Base Case: Large Light Water Reactor			
AP1000 Reactor Westinghouse Electric	3.4 GWt 1.09 GWe	Pressurized water coolant 15.5 MPa	LEU once-through 49 MWd/kgHM
Modular Light Water Reactor			
NuScale Small Modular Reactor	250 MWt 77 MWe	Natural circulation cooling 12 PWR modules	LEU once-through 41-60 MWd/kgHM
Fast-Spectrum Reactor			
Versatile Test Reactor U.S. Dept. of Energy	300 MWt 0.0 MWe	Na cooled Na-bonded fuel	U-20Pu-10Zr metallic fuel 5.0 wt% ^{235}U
Natrium TerraPower	840 MWt 339 MWe	Na cooled 1.0-m fuel column	U-10Zr HALEU 16.7 wt% 150 MWd/kgHM
ARC-100 Advanced Rx Concepts	286 MWt 100 MWe	Na cooled, superheated steam for balance of plant	HALEU metallic alloy 77 MWd/kgHM
SEALER-55 LeadCold	140 MWt 55 MWe	Lead cooled, Hf and Enriched ^{15}N in fuel	UN HALEU 12.0 wt% 60 MWd/kgHM
MCFR TerraPower	0.5–1.2 GWe	Molten chloride fuel online refueling	U-Pu HALEU 12.0 wt% 180-360 MWd/kgHM
eVinci Microreactor Westinghouse	4.5 MWe	Heat pipe fuel pin design	TRISO UCO HALEU 19.8 wt%, 45 MWd/kgHM
Aurora Powerhouse Oklo	4.0 MWt 1.5 MWe	Heat pipe metallic annular fuel	U-10Zr, HALEU 12-19.8 wt%, 20 MWd/kgHM
Lead Fast Reactor Westinghouse	950 MWt 450 MWe	Lead cooled, secondary supercritical water	UO_2 HALEU, 13.8 wt% 80 MWd/kgHM peak
Gas-Cooled Reactor			
Xe-100 X-Energy	200 MWt 80 MWe	He cooled pebble-bed reactor	TRISO UCO HALEU 15.5 wt% 160 MWd/kgHM[a]

continued

TABLE E.2 Continued

Project Description	Power	Features	Fuel Cycle
KP-X FHR Kairos Power	320 MWt 140 MWe	LiF-BeF_2 molten salt pebble-bed reactor	TRISO HALEU 19.5 wt% 40-mm pebbles, 194 MWd/kgHM
SC-HTGR Framatome	625 MWt 272 MWe	He cooled prismatic HTGR	TRISO UC HALEU 14.5 wt% 160 MWd/kgHM
BANR HTGR BWXT Technologies	50 MWt 17 MWe	He cooled prismatic HTGR	TRISO UCO HALEU 19.8 wt%
Energy Multiplier EM2 General Atomics	500 MWt 250 MWe	Pressurized He coolant	UC pellet, SiC cladding 143 MWd/kgHM
Molten Salt Fueled Reactor[b]			
IMSR-400 Terrestrial Energy	400 MWt 195 MWe	C-moderated molten salt fuel	UF_4 LEU 4.95 wt% 29 MWd/kgHM
ThorCon	557 MWt 250 MWe	Fluoride molten salt 5% fuel salt, 95% C	85% ThF_4, 15% UF_4 19.7 wt% fissile, 509 MWd/kgHM
LFTR Flibe Energy	600 MWt 250 MWe	LiF-BeF_2 molten salt	LiF-BeF_2 -^{233}U F_4, 19.6 wt% 194 MWd/kgHM
Moltex SSR-W Moltex Energy	750 MWt	KCl coolant in cylindrical fuel pins	CANDU spent fuel as feed 150 MWd/kgHM

[a] In August 2022, X-energy provided an updated burnup to the committee of 165 MWd/kg HM.

[b] Following a prepublication version of this report, the row heading was renamed to specify that the reactors in this category are both cooled and fueled by molten salt.

NOTE: CANDU = Canadian Deuterium Uranium; FHR = fluoride-cooled high-temperature reactor; GWt = gigawatt thermal; GWe = gigawatt electric; HALEU = high assay low-enriched uranium; HTGR = high-temperature gas-cooled reactor; LEU = low-enriched uranium; LFTR = liquid fluoride thorium reactor; MPa = megapascal; MWd/kgHM = megawatt-day per kilogram of heavy metal; MWe = megawatt electric; MWt = megawatt thermal; PWR = pressurized water reactor; TRISO = TRistructural ISOtropic.

MATERIAL FLOW SHEETS FOR REPRESENTATIVE FUEL CYCLES

The corresponding fuel cycle characteristics are also discussed in terms of the inventories of nuclear fuel material, referred to as "heavy metal" (HM), which comprises transuranic elements, Pu, Am, Np, and Cm, and all other actinides, usually starting with Th and U, used as fuel in nuclear reactors and those produced during the fuel cycle. Among the TRUs, Am, Np, and Cm are designated as *minor actinides* (MAs). Representative HM flow sheets are presented in Figures E.2–E.5 for three representative reactor types for once-through fuel cycles beginning with fresh fuel, together with a continuous recycling fluid system. Figure E.6 discusses a fuel recycling configuration for a combined PWR-SFR (sodium-cooled fast reactor) configuration. All of the flow sheets are normalized to the power output of 1.0 GWe.

Pressurized Water Reactor

The PWR design represents the current generation of LWRs utilizing low-enrichment uranium fuel with ^{235}U enrichment less than 5.0 wt%. The AP1000 plant, representing Generation III+ designs, features fuel assemblies with an effective length of 4.27 m (14 feet) and rated power generation of 1.1 GWe, with a heat-to-electricity conversion efficiency of 33 percent. The NuScale Small Modular Reactor design under development features 12 smaller PWR units with a fuel length reduced to 2.0 m and a rated power of 77 MWe each. The average energy of neutrons in a PWR core is approximately 0.25 eV.

A typical PWR core is loaded with 200 UO_2 fuel assemblies with a total inventory of 90 MgU or 90 MgHM at the beginning of cycle, as indicated in Figure E.2. One-third of the fuel elements with an inventory of 30 MgHM

are discharged and new fuel elements are loaded and shuffled with partially used fuel elements with a cycle length of 18 months. This explains the charge and discharge fuel inventory of 21.9 MgHM per year and the nominal fuel residence time of 4.5 years, reduced to an effective fuel residence time of 4.1 years accounting for routine maintenance. This then yields a discharge fuel burnup of 50 MWd/kgHM or 50 GWd/MTU for fuel elements operating at a power density of 3.0 GWt//90 MgHM = 33.3 kWt/kgHM. The discharged fuel elements contain 4–5 percent of FPs, 1.2 percent TRU, including 0.1 percent MA, and 94–95 percent unused uranium.

An estimate of the FP generation rate may be obtained from an approximate relationship (Lee, 2020):

> Fuel burnup [MWd/kgHM] = 939 *fima* (fissions per initial metal atom),
> with the observation that approximately 200 MeV of energy is released per fission event.

With a discharge burnup of 50 MWd/kgHM for the PWR configuration, the energy balance statement yields *fima* = 0.053 or an FP yield of 1.2 Mg/year, in agreement with the detailed *NFCE&S* calculation included in Figure E.2.

Approximately two-thirds of the 93 LWRs operating in the United States in 2022 are PWRs, and the rest of the fleet are BWRs. The fuel elements or bundles are a bit smaller in a typical BWR core, which contains a slightly larger number of fuel bundles. Thus, it is reasonable to assume that the fuel cycle characteristics of Figure E.2 are applicable also to BWR plants. Several modular LWR plants under development, including the NuScale Small Modular Reactor, feature low-enrichment uranium oxide fuel similar to that used in LWR plants and fuel cycle characteristics will not change much from those of Figure E.2, subject to possible improvements with increased thermal efficiency.

Sodium-Cooled Fast Reactor

The SFR design features one of the key Generation IV reactors fueled with uranium and TRU elements in the form of metallic U-Zr or U-Pu-Zr fuel rods. The SFR core typically features tight fuel assembly designs with the assembly length much shorter than that for LWR designs. Liquid-sodium coolant produces a hard neutron spectrum with an average neutron energy around 0.1–0.2 MeV, providing flexibility for using a variety of fuel materials including TRUs from used LWR fuel assemblies. In the fast neutron spectrum, the fission-to-capture cross section ratio for ^{235}U or ^{239}Pu is smaller than that in the thermal spectrum of the LWRs by an order of magnitude, which requires higher fissile enrichments for fast reactors, represented by the high-assay low-enriched uranium (HALEU) fuel composition planned for the demonstration core of the TerraPower Natrium reactor. This feature is discussed in connection with Table E.1.

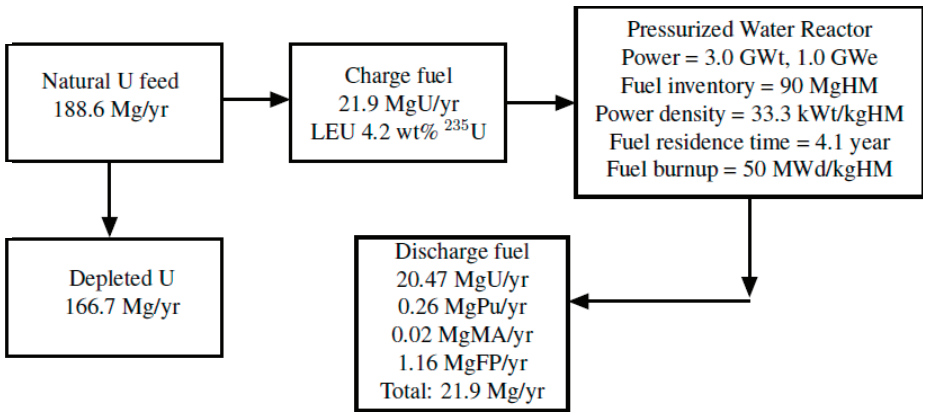


FIGURE E.2 Material flow for a pressurized water reactor design with once-through low-enrichment uranium (LEU) fuel. SOURCE: Adapted from NFCE&S Evaluation Group 1, Wigeland et al. (2014).

The higher average energy of the fast-spectrum reactors would allow efficient production of fissile Pu from ^{238}U and could work efficiently with the MAs Am, Np, and Cm, allowing for the breeding of fissile materials for subsequent fast reactor cycles or as feed material for LWRs. With the HALEU fuel with 16.7 wt% fissile ^{235}U enrichment, the charge fuel inventory per year and HM inventory in the Natrium Advanced Type 1B Metal core, summarized in Figure E.3, are both reduced significantly from the PWR counterparts in Figure E.2. TerraPower plans to complete the development of fuel element designs without sodium bonding for the Type 1B Metal Core. With an increased thermal efficiency, the Natrium core design, as a fast-spectrum reactor, indicates a three-fold increase in the discharge fuel burnup and a four-fold reduction in the used nuclear fuel inventory discharged, compared with an equivalent 1.0-GWe LWR plant. It should, however, be noted that the depleted U inventory increases by ~15 percent from the LWR counterparts.

High-Temperature Gas-Cooled Reactor

The design utilizes submillimeter diameter TRistructural ISOtropic (TRISO) particles with multiple pyrolytic graphite coatings that form the basic building block. The prismatic design loads TRISO particles, often featuring uranium oxycarbide (UCO) fuel, into graphite pin cells or compacts, while in the pebble-bed design the TRISO particles are packed into graphite spheres with a diameter of 40–60 mm. The high-temperature gas-cooled reactor (HTGR) designs provide mostly thermal spectrum but may be configured to provide an epithermal spectrum with an average neutron energy in the eV range.

The fuel characteristics summarized in Figure E.4 illustrate a HALEU design with a fissile U enrichment of 15.5 wt% in a thermal neutron spectrum of a graphite-moderated HTGR, similar to the Xe-100 pebble-bed reactor design. Utilizing a HALEU design allows a charge and discharge fuel rates similar to those for the HALEU-fueled SFR of Figure E.3. With a high He coolant temperature possible with the HTGR, a higher thermal efficiency of 50 percent could be attained, compared with 41 percent for the Natrium design. With a higher power density of 67.1 kWt/kgHM, compared with 41.3 kWt/kgHM for the Type 1B Natrium core, a discharge burnup of 120 MWd/kgHM could be obtained for the HTGR in 4.9 years compared with a discharge burnup of 150 MWd/kgHM in 10 years for the Natrium design. The FP production rate is reduced somewhat also for the HTGR design with a higher thermal efficiency compared with the SFR design.

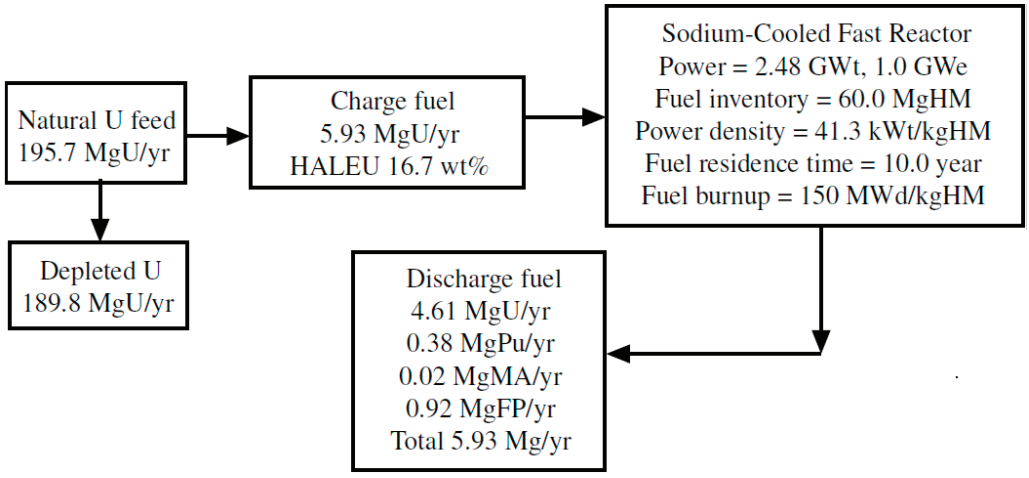


FIGURE E.3 Material flow for a sodium-cooled fast reactor design with once-through high-assay low-enriched uranium fuel (HALEU).
SOURCE: Adapted from TerraPower communication, dated January 14, 2022. Copyright © 2022 TerraPower, LLC. All rights reserved.

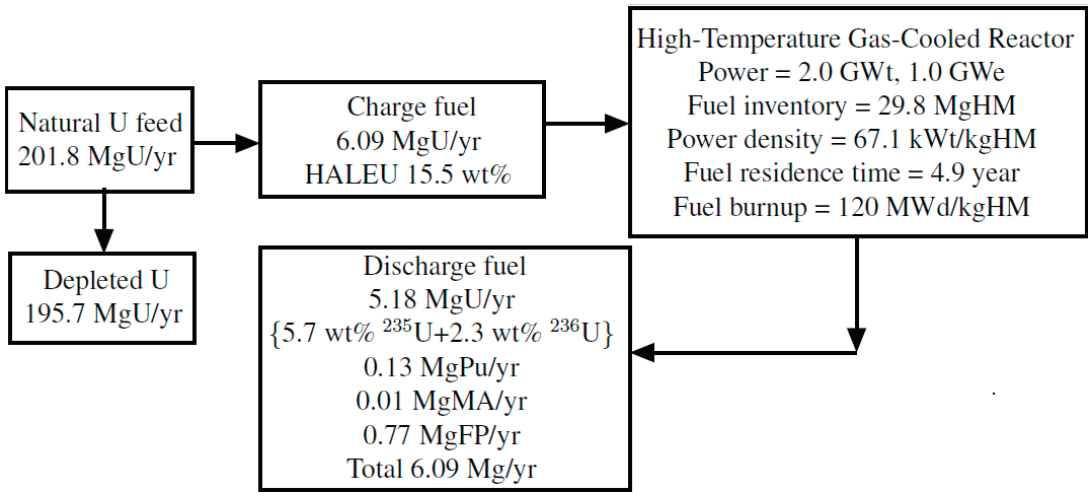


FIGURE E.4 Material flow for a high-temperature gas-cooled reactor design with high-assay low-enriched uraniuim (HALEU) TRistructural ISOtropic (TRISO) fuel.
SOURCE: Adapted from NFCE&S Evaluation Group 2, Wigeland et al. (2014).

Molten Salt Reactor

The molten salt reactor (MSR) design illustrates the use of various salts, including LiF-BeF_2 molten fluoride salt, as a coolant for the core, as well as the salt admixed with fuel. The material flow sheet in Figure E.5 represents a molten fuel design that allows continuous circulation of the fuel salt with an equivalent period of 3 days, or 121.67 times per year, indicating a large flow rate of fuel salt summed over a year as high as 8,862 Mg/year. The fuel residence time of 76.5 years is estimated from the limited recycle MSR case, represented as Evaluation Group 10, and essentially indicates that the fuel salt circulates through the system indefinitely, resulting in a large fuel burnup.

The MSR system provides a continuous online fuel salt treatment with low processing loss so that the material processed and discharged consists primarily of FPs produced during the reactor operation. The circulating salt comprises Th as the feed material producing ^{233}U as the fissile material demonstrating the significant potential of a Th converter operating with merely 0.93 Mg/year of Th as feed material; this may be compared with 21.9 Mg/year of low-enrichment uranium (LEU) fuel for the once-through PWR cycle in Figure E.2. In the thermal-spectrum MSR design featuring a Th-U fuel cycle in Figure E.5, the production of TRU elements is very small, so that the sum of the ^{233}U, Th, and MA inventories discharged through the fuel salt treatment is merely 0.02 Mg/year. The fissile U content, comprising ^{233}U and ^{235}U, is 69.0 wt% in the recirculating fuel stream, and may require denaturing the fuel cycle with the addition of ^{238}U.

Depending on the salt utilized, an MSR could be configured either as a thermal or fast-spectrum reactor. With KCl salt as the coolant and $UPuCl_3$ as the fuel, the Moltex design indeed suggests a fast-spectrum reactor configuration with an online refueling strategy and spent Canadian Deuterium Uranium (CANDU) fuel used as the feed material.

Two-Tier Coupled SFR-PWR System

Coupling an SFR with a full breeding capability to a PWR in a symbiotic manner, the system could recycle the Pu generated in both the PWR and SFR cores. The flow sheet of Figure E.6 represents a synergistic arrangement where only Pu, together with the uranium fuel, is recycled and MAs and FPs are sent to a repository. The only feed material required then is a small amount of natural uranium for the combined system replenishing the FPs and MAs to be sent to a repository.

The coupled system is structured so that the combined power output is 1.0 GWe, with a 60%–40% split between

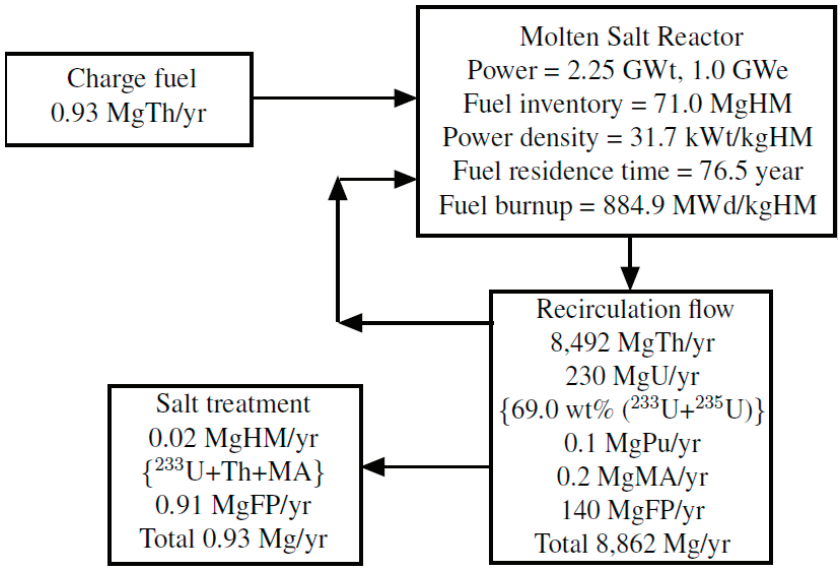


FIGURE E.5 Material flow for a molten salt reactor design with circulating high-assay low-enriched uranium (HALEU) and thorium fuel.
SOURCE: Adapted from NFCE&S Evaluation Group 26, Wigeland et al. (2014).

the SFR and the PWR. For the SFR itself, the power production is split 90%–10% between the driver and blanket regions. The PWR configuration is nearly equivalent to that of Figure E.2, if the power output is normalized to 1.0 GWe, with the fuel residence time of 3.9 year compared with 4.1 year in Figure E.2 and the identical discharge fuel burnup of 50 MWd/kgHM. The main difference, of course, is the PWR use of Pu remaining from the previous PWR cycle combined with Pu produced in the SFR. With full recycling, the inventory of U and Pu from the previous SFR cycle is split between the charge to the SFR and additional mix to the PWR to bring up the required fissile contents for the PWR charge fuel. It should be also noted that the discharged fuel from both the SFR and PWR comprise MA and FP, and a total of 0.23 Mg of inventory, out of a total feed inventory of 1.33 Mg of natural U (i.e., equivalent to 17 percent of the total feed stream, is assumed lost during the recycling). Furthermore, it is useful to note that the 60%–40% power split between the SFR and PWR is approximately represented in the FP production split of 0.56 Mg versus 0.43 Mg. The total discharge fuel inventory, including the reprocessing loss, is naturally equal to the total feed rate of 1.33 Mg/year, which compares with the discharge rates of 21.9 Mg/year for the open-cycle PWR in Figure E.2 and 5.9 Mg/year for the open-cycle SFR in Figure E.3.

Two important final observations relate to (a) net Pu *production* of (1.09–0.85 = 0.24) Mg in the SFR versus net Pu *consumption* of (0.79–0.54 = 0.25) Mg in the PWR and (b) MA production of 0.02 Mg in the SFR versus 0.07 Mg in the PWR, despite the 60%–40% power split between the two reactors. These observations translate to a key potential of the SFR serving as a breeder of fissile Pu with a minor negative impact due to tracer quantities of MA present in the recycled fuel. These features are not explicitly illustrated in the Natrium design summarized in Figure E.3, because the design features U-Zr HALEU fuel.

COMPARISON OF MATERIAL FLOW AND REPOSITORY METRICS

The charge and discharge fuel material flows illustrated in Figures E.2–E.6 are summarized in Table E.3 and compared to clarify the characteristics of the five fuel cycles chosen to represent advanced reactor designs evaluated. The repository metrics including radioactivity at 100 and 100,000 years are compared for the three advanced reactor designs, together with the reference PWR and two-tier SFR-PWR recycle system. In the *NFCE&S* report, an adjustment is made to the discharge fuel inventories and the corresponding repository metrics to normalize the thermal efficiency to 33 percent selected for the reference PWR design (Wigeland et al., 2014). For the analysis of repository

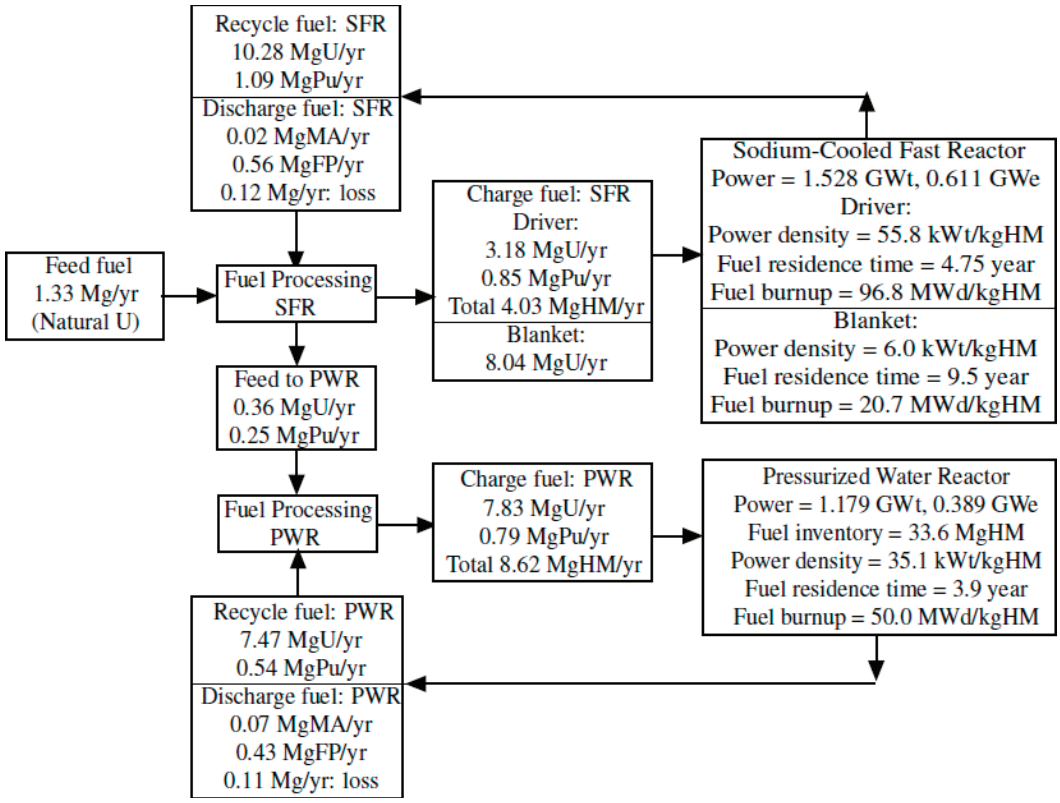


FIGURE E.6 Material flow for a coupled SFR-PWR system, utilizing Pu-U-Zr fuel for the SFR and U-Pu mixed oxide fuel for the PWR.
SOURCE: Adapted from NFCE&S Evaluation Group 29, Wigeland et al. (2014).

metrics, the committee decided not to adopt the normalization process, because the increased thermal efficiencies of the advanced reactor designs represent a unique feature of the advanced concepts and hence should be retained.

For LWRs with ~33% thermal efficiency, FP inventories reflect a simple rule of thumb that a 1.0-GWe nuclear power plant consumes approximately 1.0 Mg/year of fissionable material. Thus, 1.0-GWe plant produces approximately 1.0 Mg of FPs in one year, as quantified via *fima*. (The FP inventories are reduced for designs with higher thermal efficiency.) The yields for short- and long-lived FPs of interest in used fuel management and geologic repository analysis depend somewhat on the neutron flux spectrum and fissile nuclides, but the sum of FP yields of interest appear nearly independent of the fuel cycle.[1]

Table E.3 summarizes the results of spent nuclear fuel and high-level waste (HLW) inventory in terms of its mass, radioactivities at 100 years and 100,000 years after discharge. These data are normalized per energy generated in order to have fair comparisons between different types of reactors. The time 100 years is selected to represent requirements for handling/storage of spent nuclear fuel and corresponding heat generation for disposal engineering designs in the geologic repository. Geologic repository loading is often limited by decay heat. While not an exact surrogate, the radioactivity and decay heat at 100 years provide a relative indication of the challenge of disposal of such waste.

Similarly, a time of 100,000 years after discharge is selected to represent the long-term isolation challenges in a deep geologic repository. The activity of spent nuclear fuel (SNF)/HLW at 100,000 years per energy generated represents the long term hazard of the SNF/HLW. It represents the total "hazard source term" of the inventory without considering the accessibility of the hazard. Tables E.4 and E.5 present the key contributing radionuclides for each of the advanced reactor types.

[1]The paragraph was modified following a prepublication version of the report to clarify that the rule of thumb applies only to light water reactors with 33 percent thermal efficiency, and that higher thermal efficiency designs will have reduced fission product inventories.

TABLE E.3 Spent Nuclear Fuel Inventory and Repository Metrics of Advanced Nuclear Reactors

Fuel Cycle and Reactor Type (Reference)	Spent Nuclear Fuel Inventory (Mg/GWe-yr)	Radioactivity at 100 years (MCi/GWe-yr)	Radioactivity at 10^5 years (kCi/GWe-yr)
Light water reactor LEU PWR (NFCE&S EG01)	21.92	1.34	1.65
Fast-spectrum reactor HALEU (TerraPower Natrium)	5.93	0.53	1.58[a]
Gas-cooled reactor HALEU TRISO (NFCE&S EG02)	6.09	0.94	1.35
Molten salt reactor HALEU Th-233U (NFCE&S EG26)	0.93	0.96	1.02
Two-tier SFR-PWR Recycle (NFCE&S EG29)	1.33	1.01	0.82

[a] Excludes contributions from isometric.

NOTE: HALEU = high-assay low-enriched uranium; LEU = low-enriched uranium; NFCE&S = nuclear fuel cycle evaluation and screening; PWR = pressurized water reactor; SFR = sodium-cooled fast reactor; TRISO = TRistructural ISOtropic.

TABLE E.4 Spent Nuclear Fuel and High-Level Waste Activities at 100 Years (kCi/GWe-yr)

PWR (EG01)		SFR (Natrium)		HTGR (EG02)		MSR (EG26)		SFR-PWR (EG29)	
Total:	1,340	Total:	530	Total:	940	Total:	959	Total:	1,010
^{137}Cs	341	^{137}Cs	262	^{137}Cs	231	^{137}Cs	245	^{137}Cs	292
^{137}Ba*	322	^{90}Sr	168	^{137}Ba*	219	^{90}Sr	235	^{137}Ba*	277
^{90}Y	215	^{238}Pu	30	^{90}Sr	161	^{90}Y	235	^{241}Am	167
^{90}Sr	215	^{239}Pu	21	^{90}Y	161	^{137}Ba*	231	^{90}Y	99
^{241}Am	122	^{240}Pu	8	^{241}Am	87			^{90}Sr	99

NOTE: HTGR = high-temperature gas-cooled reactor; MSR = molten salt reactor; PWR = pressurized water reactor; SFR = sodium-cooled fast reactor; * indicates nucleus in an excited state.

TABLE E.5 Spent Nuclear Fuel and High-Level Waste Activities at 100,000 Years (Ci/GWe-yr)

PWR (EG01)		SFR (Natrium)		HTGR (EG02)		MSR (EG26)		SFR-PWR (EG29)	
Total:	1,650	Total:	1,580	Total:	1,350	Total:	1,020	Total:	818
^{239}Pu	494	^{239}Pu	1,180	^{239}Pu	228	^{99}Tc	245	^{99}Tc	289
^{99}Tc	304	^{99}Tc	254	^{99}Tc	210	^{93}Zr	62	^{239}Pu	137
^{242}Pu	61	^{93}Zr	48	^{234}U	77	^{93}Nb*	59	^{233}Pa	43
^{93}Zr	56	^{135}Cs	40	^{222}Rn	52	^{209}Pb	53	^{237}Np	43
^{93}Nb*	54	^{234}U	20	^{218}Po	52	^{213}Bi	53	^{135}Cs	39
^{234}U	53	^{237}Np	12	^{226}Ra	52	^{217}At	53	^{93}Zr	36
^{233}Pa	37	^{236}U	10	^{214}Pb	52	^{221}Fr	53	^{93}Nb*	35
^{237}Np	37			^{214}Bi	52	^{229}Th	53	^{126}Sb*	16
^{222}Rn	35			^{210}Po	52	^{225}Ra	53	^{126}Sn	16
^{214}Pb	35			^{210}Pb	52	^{225}Ac	53	^{233}U	15
^{218}Po	35			^{214}Po	52	^{213}Po	51	^{213}Bi	14
^{214}Bi	35			^{210}Bi	52	^{233}U	50	^{217}At	14
^{226}Ra	35			^{230}Th	52	^{234}U	16	^{221}Fr	14
^{210}Pb	35			^{242}Pu	43	^{126}Sb*	16	^{225}Ra	14
^{210}Po	35			^{93}Zr	38	^{126}Sn	16	^{225}Ac	14
^{210}Bi	35			^{93}Nb*	37	^{210}Po	11		
^{214}Po	35			^{233}Pa	27	^{214}Po	11		
^{230}Th	35			^{237}Np	27	^{222}Rn	11		
^{135}Cs	15			^{135}Cs	17				
^{233}U	13								

NOTE: HTGR = high-temperature gas-cooled reactor; MSR = molten salt reactor; PWR = pressurized water reactor; SFR = sodium-cooled fast reactor; * indicates nucleus in an excited state.

Key general observations are offered:

1. With the fission process releasing two FPs and approximately 200 MeV of energy per fission, a 1.0-GWe power plant consumes approximately 1.0 Mg/year of nuclear fuel and produces 1.0 Mg/year of FPs, making the FP inventory in UNF proportional to the thermal energy generated. The discharge fuel and FP inventories are thus reduced for designs with a higher thermal efficiency.
2. Effective use of HALEU with a thermal efficiency of 40–50 percent for the SFR and HTGR, compared with 33 percent for the PWR, reduces the fuel inventory to 25 percent of the PWR inventory. The continuous circulating MSR and coupled SFR-PWR system requiring fuel reprocessing reduce the UNF inventory to 5 percent of the PWR discharge fuel.
3. On a per-energy-generation basis, radioactivity at 100,000 years, with a significant portion from actinides, is reduced in SFR from a few percent below the level of PWR to somewhat more than a factor of 2 for SFR-PWR as compared to PWR.[2] These reductions do not have, however, significant impacts on the safety of a repository if geochemical and geologic conditions of the repository site are carefully chosen to limit the mobility and accessibility of the actinides.

[2] This sentence was modified following a prepublication version of the report to clarify the comparison between radioactivity at 100,000 years for these reactor designs.

4. Together with a priority given for the development of a geologic repository for the legacy UNF, additional effort effectively utilizing fuel reprocessing could help reduce the future accumulation of UNF from advanced reactors under development.
5. The coupled SFR-PWR system features a net production of Pu in the SFR, which is consumed in the PWR. The production of MAs is significantly smaller in the SFR than in the PWR. These observations translate to a key potential of the SFR serving as a breeder of fissile Pu with a minor negative impact due to tracer quantities of MAs in the recycled fuel.

Appendix F

Sample List of National Academies Reports on Nuclear Waste Management

NRC (National Research Council). 2011. *Waste Forms Technology and Performance: Final Report*. https://doi.org/10.17226/13100.

NRC. 2010. *Science and Technology for DOE Site Cleanup: Workshop Summary*. https://doi.org/10.17226/11932.

NRC. 2009. *Advice on the Department of Energy's Cleanup Technology Roadmap*. https://doi.org/10.17226/12603.

NRC. 2006. *Tank Waste Retrieval, Processing, and On-Site Disposal at Three Department of Energy Sites: Final Report*. https://doi.org/10.17226/11618.

NRC. 2005a. *Improving the Characterization and Treatment of Radioactive Wastes for the Department of Energy's Accelerated Site Cleanup Program*. https://doi.org/10.17226/11200.

NRC. 2005b. *Risk and Decisions about Disposition of Transuranic and High-Level Radioactive Waste*. https://doi.org/10.17226/11223.

NRC. 2004. *Improving the Characterization Program for Contact-Handled Transuranic Waste Bound for the Waste Isolation Pilot Plant*. https://doi.org/10.17226/10900.

NRC. 2003a. *End Points for Spent Nuclear Fuel and High-Level Radioactive Waste in Russia and the United States*. https://doi.org/10.17226/10667.

NRC. 2003b. *Improving the Scientific Basis for Managing DOE's Excess Nuclear Materials and Spent Nuclear Fuel*. https://doi.org/10.17226/10684.

NRC. 2003c. *Long-Term Stewardship of DOE Legacy Waste Sites: A Status Report*. https://doi.org/10.17226/10703.

NRC. 2002a. *Characteristics of Remote-Handled Transuranic Waste at the Waste Isolation Pilot Plant*. https://doi.org/10.17226/10492.

NRC. 2002b. *Research Opportunities for Managing the Department of Energy's Transuranic and Mixed Waste*. https://doi.org/10.17226/10513.

NRC. 2001a. *Research Needs for High-Level Waste Stored in Tanks and Bins at U.S. Department of Energy Sites: Environmental Management Science Program*. https://doi.org/10.17226/10191.

NRC. 2001b. *Research Opportunities for Deactivating and Decommissioning Department of Energy Facilities*. https://doi.org/10.17226/10184.

NRC. 2001c. *A Strategic Vision for Department of Energy Environmental Quality Research and Development*. https://doi.org/10.17226/10207.

NRC. 2000. *Long-Term Institutional Management of U.S. Department of Energy Legacy Waste Sites*. https://doi.org/10.17226/9949.

NRC. 1999a. *Alternative High-Level Waste Treatments at the Idaho National Engineering and Environmental Laboratory*. https://doi.org/10.17226/9743.

NRC. 1999b. *Decision Making in the U.S. Department of Energy's Environmental Management Office of Science and Technology*. https://doi.org/10.17226/9448.

NRC. 1999c. *Peer Review in Environmental Technology Development Programs: The Department of Energy's Office of Science and Technology*. https://doi.org/10.17226/6408.
NRC. 1999d. *The State of Development of Waste Forms for Mixed Wastes: U.S. Department of Energy's Office of Environmental Management*. https://doi.org/10.17226/9459.
NRC. 1999e. *Technologies for Environmental Management: The Department of Energy's Office of Science and Technology*. https://doi.org/10.17226/9656.
NRC. 1998a. *A Review of Decontamination and Decommissioning Technology Development Programs at the Department of Energy*. https://doi.org/10.17226/6290.
NRC. 1998b. *Study of the Decision Process Related to Long-Term Disposition of U.S. Department of Energy Waste Sites and Facilities: Interim Report*.
NRC. 1997a. *Barrier Technologies for Environmental Management*. https://doi.org/10.17226/5539.
NRC. 1997b. *Building an Effective Environmental Management Science Program: Final Assessment*. https://doi.org/10.17226/5557.
NRC. 1997c. *Evaluation of the U.S. Department of Energy's Alternatives for the Removal and Disposition of Molten Salt Reactor Experiment Fluoride Salts*. https://doi.org/10.17226/5538.
NRC. 1996a. *Affordable Cleanup? Opportunities for Cost Reduction in the Decontamination and Decommissioning of the Nation's Uranium Enrichment Facilities*. https://doi.org/10.17226/5114.
NRC. 1996b. *Glass as a Waste Form and Vitrification Technology: Summary of an International Workshop*. https://doi.org/10.17226/5488.
NRC. 1996c. *The Waste Isolation Pilot Plant: A Potential Solution for the Disposal of Transuranic Waste*. https://doi.org/10.17226/5269.
NRC. 1995. *Safety of the High-Level Uranium Ore Residues at the Niagara Falls Storage Site, Lewiston, New York*. https://doi.org/10.17226/9161.
NRC. 1994. *Building Consensus Through Risk Assessment and Management of the Department of Energy's Environmental Remediation Program*. https://doi.org/10.17226/9480.
NRC. 1992. *Review of Analyses by the U.S. Department of Energy of Selected Technical Issues in the Environmental Protection Agency Standards for High-Level and Transuranic Radioactive Wastes* (40 CFR 191).
NRC. 1991. *WIPP Panel Letter Report Summarizing Views on the Status of DOE's Scientific and Technical Program to Assess the WIPP's Ability to Isolate Transuranic (TRU) Waste and to Demonstrate Compliance with the Relevant Regulations*.
NRC. 1989. *WIPP Panel Letter Report of Review Comments on DOE Document DOE/WIPP 89-011: Draft Plan for the Waste Isolation Pilot Plant Test Phase: Performances Assessment and Operations Demonstration*.
NRC. 1988a. *WIPP Panel Letter Report on Brine Accumulation in the WIPP Facility*.
NRC. 1988b. *WIPP Panel Letter Report on Intermediate-Scale Experiments of Room Closure Rates*.
NRC. 1987a. *BRWM Letter Report on the Defense Waste Management Plan for Buried Transuranic Waste, Transuranic Contaminated Soil, and Difficult-to-Certify Transuranic Waste*.
NRC. 1987b. *Scientific Basis for Risk Assessment and Management of Uranium Mill Tailings*. https://doi.org/10.17226/18922.
NRC. 1987c. *WIPP Panel Letter Report on the Planned Sorbing Tracer Tests at the Waste Isolation Pilot Plant*.
NRC. 1984a. *Review of the Scientific and Technical Criteria for the Waste Isolation Pilot Plant*.
NRC. 1984b. *Social and Economic Aspects of Radioactive Waste Disposal*. https://doi.org/10.17226/316.
NRC. 1983. *Review of the Criteria for the Site Suitability, Design, Construction, and Operation of the Proposed Waste Isolation Pilot Plant (WIPP): Interim Report*.
NRC. 1980. *Continuing Evaluation of the Carlsbad Site: Report to the U.S. Department of Energy*.
NRC. 1979a. *Implementation of Long-Term Environmental Radiation Standards: The Issue of Verification*. https://doi.org/10.17226/19819.
NRC. 1979b. *Solidification of High-Level Radioactive Wastes*.
NRC. 1978. *Geological Criteria for Repositories for High-Level Radioactive Waste*.
NRC. 1974. *Transportation of High-Level Nuclear Wastes: Letter Report of the Panel on Transportation of Radioactive Wastes*.
NRC. 1972. *An Evaluation of the Concept of Storing Radioactive Waste in Bedrock Below the Savannah River Plant Site*.
NRC. 1970a. *Disposal of Solid Radioactive Waste in Bedded Salt Deposits*.
NRC. 1970b. *Radioactive Waste Management: An Interim Report of the Committee on Radioactive Waste Management*. https://doi.org/10.17226/21764.
NRC. 1966. *Review of Radioactive Waste Disposal Technology—Report to the Division of Reactor Development and Technology, U.S. Atomic Energy Commission*.
NRC. 1958. *Thermal Considerations in Deep Disposal of Radioactive Waste*.
NRC. 1957. *The Disposal of Radioactive Waste on Land*. https://doi.org/10.17226/10294.

Appendix G

Reprocessing and Geologic Disposal of TRISO Fuel

TRISO FUEL REPROCESSING

Several advanced reactor designs intend to use TRistructural ISOtropic (TRISO) fuel, which is attractive for its robustness and ability to withstand high temperatures. However, these properties also make processing the fuel to access and recover fissile material a significant challenge, so most research has focused on developing head-end steps. In general, processing of TRISO fuel elements consists of the following steps: "mechanical preparation, removal of carbon external to the SiC shell, removal of the SiC shell, removal of carbon layers between the SiC shell and heavy metal kernel, and dissolution of the heavy metal kernel" (Del Cul et al., 2002). Initial work in TRISO fuel processing involved burning the outer layers of graphite and then crushing the fuel particles using steel rollers to expose the fuel for dissolution in nitric acid, followed by processing using the standard PUREX flow sheet (IAEA, 2008). While effective, this method releases CO_2 containing quantities of ^{14}C above regulatory limits, and therefore must be paired with an expensive off-gas treatment system and sequestration methods that generate significant quantities of calcium carbonate waste (Del Cul et al., 2002). Consequently, alternative strategies that simplify processing steps, minimize waste, and can take advantage of existing industrial processes and equipment are the focus of current research (Del Cul et al., 2002).

Methods have been developed to recover fissile material from TRISO fuel in forms suitable for both aqueous and nonaqueous separations. The initial steps are the same in both cases and involve crushing and milling the fuel compacts into fine particle size. The fuel is then dissolved by nitric acid leaching for aqueous processing, or by carbochlorination for nonaqueous processing (Del Cul et al., 2002). An alternative approach under development uses supercritical CO_2 containing tributylphosphate to extract uranium from crushed TRISO particles (Zhu et al., 2012). Other nonaqueous options include fluoride and chloride volatility and direct electrochemical dissolution (IAEA, 2008).

Savannah River National Laboratory has developed and patented a molten salt dissolution process in which the outer graphite layer of the TRISO fuel is oxidized by a nitrate salt at temperatures between 400 and 700°C, generating carbon dioxide, carbonate salts, nitrous acid, and nitrogen (Pierce, 2017). Addition of an alkali metal hydroxide, either in the same or a subsequent step, oxidizes the silicon carbide layer, and residual nitrate salt then oxidizes the inner carbon layers that surround the fuel kernel. Finally, a metal peroxide or superoxide is used to solubilize the fuel kernel. Nitric acid is also required for this dissolution step if the fuel contains thorium oxide. The overall process can occur in one step, using a single processing vessel, or by a multistep sequential method. For the former, mechanical pretreatment of the TRISO particles is typically required to expose the inner layers of

the fuel element so that all chemical reactions can occur simultaneously. The multistep method allows the products of graphite layer oxidation to be separated out as low-level waste.

Remaining challenges for TRISO fuel reprocessing include handling the volatile fission products in the off-gas, improving the recovery efficiencies, and scaling up all steps of the process. Nitric acid leaching can leave solid residues, such as undissolved noble metals, SiC shell fragments, SiO_2, and $NaSiO_3$-containing residues, which can lead to the formation of silicic acid and complicate subsequent separation steps (Del Cul et al., 2002). Handling and disposing of the large amount of ^{14}C-contaminated graphite presents another challenge, as discussed further in Chapter 5. Developing improved methods for removing the graphite layers surrounding the fuel kernel could enable their disposal as low-level waste rather than high-level waste (Todd, 2020).

TRISO FUEL FORM AND WASTE DISPOSAL

As one of the most well-studied of the advanced fuel forms, TRISO particles serve as a useful example of the type of attention (research and development) required in order to qualify a fuel for disposal. The United States already has TRISO-based fuels that require storage and disposal at the Fort St. Vrain (FSV) reactor site in northeast Colorado, and the challenges in dealing with that spent fuel can provide valuable lessons for future management of advanced fuel forms. A number of studies, in the United States and worldwide, have analyzed requirements for storage of spent TRISO fuel, direct disposal of spent TRISO fuel in a repository, and processing and treatment options for spent TRISO fuel prior to disposal. From these studies, additional research needs for identifying safe methods for disposing of spent TRISO fuel can be identified. More generally, the type and extent of research and testing needed to understand in-reactor behavior and analyze disposal options for TRISO fuel demonstrate the amount of work required to be able to safely manage and dispose of any advanced fuel form. In-reactor testing of new fuels can take several years.

The inventory of spent TRISO fuel in the United States was generated by the operation of a high-temperature, gas-cooled reactor by the Public Service Company of Colorado that ran commercially from 1979 to 1989, generating 23.35 MTHM (metric tons of heavy metal). Over a third of this fuel was transferred to Idaho National Laboratory (INL), but in 1989, the state of Idaho blocked further shipments to INL. Thus, the Public Service Company of Colorado constructed an independent spent fuel storage facility at the reactor site. The balance of the fuel has been transferred to the storage facility. In the mid-1990s, the U.S. Department of Energy (DOE) took possession of the fuel storage facility. The fate of the fuel is constrained by legal agreements between DOE and the states of Idaho and Colorado. The 1995 Settlement Agreement prevents the transfer of the fuel from FSV until a permanent geologic repository or interim storage facility is established outside of Idaho (IDEQ, 1995). By 2035, all DOE spent fuel must be removed from Idaho. The agreement also limits the total quantity of spent fuel (16 MTHM) that can be shipped to Idaho. In an agreement between the state of Colorado and DOE, DOE must remove all of the spent fuel from FSV by 2035 or pay a penalty. This history of the stranded TRISO fuel at FSV illustrates the larger challenges of dealing with nuclear fuels stranded at reactor sites (see NWTRB, 2017).

Radiation Effects on TRISO Fuel

As with any new nuclear fuel type, TRISO particles must undergo in-reactor testing to understand radiation effects on the fuel material. The source term of the fuel material is the starting point of any safety assessment. It is especially important that this source term characterize the redistribution of and phase formation of radionuclide phases as they are generated during reactor operating conditions. In a very complete study, Gerczak et al. (2020) examined 72 cylindrical compacts, each with 4,100 TRISO kernels. Irradiation of the compacts for ~620 full-power days to burnups of 11 to nearly 20 percent fissions per initial heavy metal content showed no indication of TRISO-layer failure during the in-pile irradiation. Concerning radionuclide distributions, the authors noted "high-Z features" that accumulated at the I[inner]PyC/SiC interface, which was interpreted as the interaction of fission products and actinides with the TRISO layers. There was also evidence of increased susceptibility of the SiC layer to decomposition at the SiC/O[outer]PyC interface. Another detailed study (van Rooyen et al., 2018) made atomic-scale observations of SiC as a function of dose using transmission electron microscopy of irradiated TRISO

fuel particles. Principal observations included (1) "black" spots, probably a mix of smaller dislocation loops; (2) polygonal voids; and (3) Frank loops, which act as nucleation sites for intergranular alpha-SiC and Pd-silicide precipitates. Van Rooyen et al. (2018) also found that the void distribution is not uniform and bimodal, and a high concentration of smaller voids occur at stacking faults. Furthermore, the retention of radionuclides appeared to be inversely proportional to void size.

Direct Disposal of TRISO in a Geologic Repository

There have been many studies relevant to the direct disposal of very-high-temperature TRISO fuels in a geologic repository, with most analyses examining the properties and performance of the graphite matrix and SiC coating within a geologic disposal environment. In general, because of their multibarriers within the waste form and matrix (as depicted in Figures G.1 and G.2), high-temperature reactor (HTR) fuel elements are considered "well designed" for direct disposal in a geologic repository (Fachinger et al., 2006).

Studies investigating properties of the graphite matrix have shown that it limits the amount of water that may come into contact with the fuel kernels (Fachinger et al., 2006). Because the embedding matrix will strongly delay groundwater contact with the fuel particles, radiolysis is probably not an important factor in the determination of fuel dissolution rates (Grambow et al., 2010). "Radionuclide transport in the water-filled graphite matrix porosity is controlled mainly by diffusion with calculated breakthrough times over a distance of 1 cm ranging from a few days up to a year" (Fachinger et al., 2006). The corrosion of matrix graphite has been investigated in water, as well as in concentrated magnesium and sodium solutions, and significant graphite corrosion was observed only

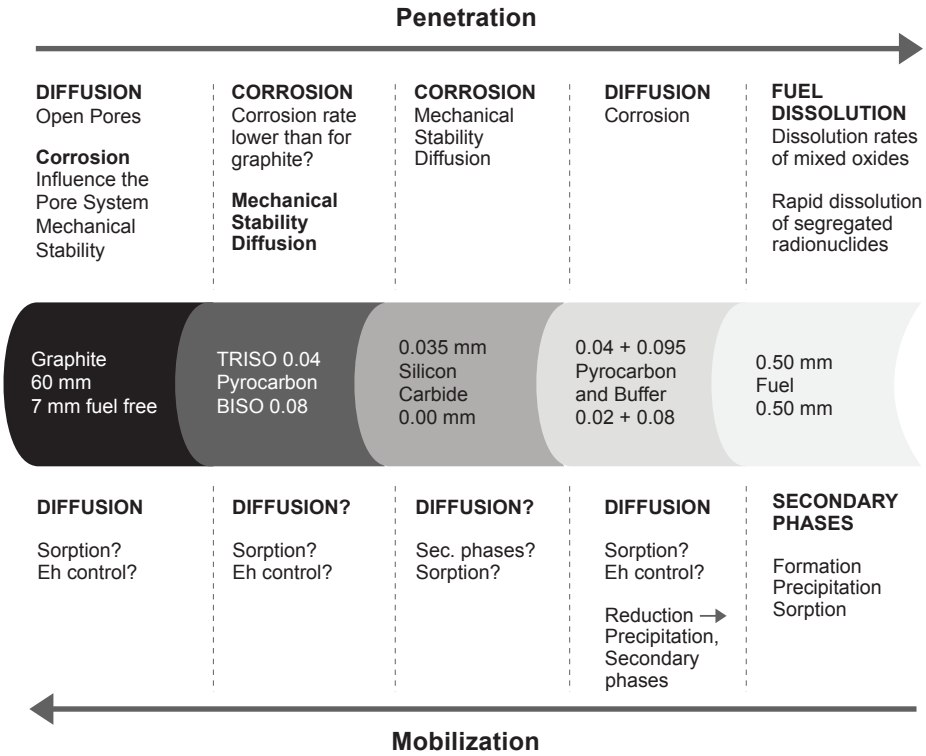


FIGURE G.1 Penetration of leachants and release of radionuclides for a spent high-temperature reactor TRistructural ISOtropic (TRISO) fuel pebble. The question marks signify issues for which additional research and development is required. SOURCES: Adapted from Fachinger et al. (2006); Grambow et al. (2010).

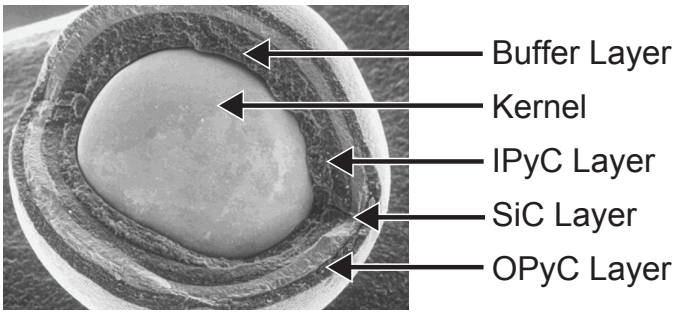


FIGURE G.2 Cross section of TRistructural ISOtropic (TRISO) fuel particle, showing fuel kernel encased in buffer, pyrolytic carbon (IPyC and OPyC), and silicon carbide (SiC) layers.
SOURCE: Idaho National Laboratory.

in the salt brines (Grambow et al., 2008). The dissolution rate in the salt brines showed a linear relation to dose, and at the highest dose (3.5 MGy), the rate was 1 mm/million years.

The properties and performance of SiC or other carbides are central to the overall performance of TRISO spent fuel in a geologic repository. Assuming that nonuniform corrosion does not occur, lifetimes of the pyrocarbon and SiC layers are estimated at 10^3–10^5 years depending on corrosion conditions, irradiation doses, and temperature; however, this requires confirmation by experimental studies (Fachinger et al., 2006). The coatings are thought to make spent TRISO fuel more resistant than conventional high-level waste forms to rapid, initial radionuclide release, based on the assumptions of the diffusion of water into the pebble in a few years, the failure of the SiC coating in 10,000 years, and the dissolution of kernels in approximately 10^6 years under reducing and 10^4 years under oxidizing conditions (Grambow et al., 2008, 2010). Studies of radiation effects (Katoh et al., 2012) and on the diffusion of fission product elements through the SiC layer (Malherbe, 2013) have provided a fundamental basis for understanding the performance of SiC. In a model that treats the SiC layer as analogous to a pressure vessel that retains gases produced by fission and radioactive decay, Peterson and Dunzik-Gougar (2011) "demonstrated that corrosion rates, temperature evolution over time and the thickness of the OPyC and SiC layers have a significant effect on the estimated time to particle failure. The dimensions of the kernel, buffer and IPyC layers along with the strength of the SiC layers and the pressure in the TRISO particle did not significantly alter the time to particle failure." The mechanical properties of TRISO fuel are also an essential aspect of their performance, and new techniques such as X-ray tomography can be used to develop an advanced understanding of these properties, particularly as a function of radiation type and dose (Liu et al., 2020).

Additional studies have examined the potential for radionuclide release from TRISO particles. Leaching tests using granite water, clay-pore water, and a Q-brine have been performed on irradiated spherical fuel elements to simulate repository conditions. These experiments showed two different phenomena with the respect to the retention of ^{137}Cs and ^{154}Eu: (1) intact fuel kernel particles retain fission products nearly completely, and (2) releases from defective or failed particles and from the matrix sphere are rapid, but the contribution to released radioactivity is small. Given the long lifetimes of SiC layers noted above, the coating should be sufficient to retain the major inventory of fission product elements that will decay to near-zero values within the first 1,000 years (Nabielek et al., 2009). Additionally, very high temperatures (1,800°C) over extended periods (650 hours) have confirmed that SiC maintained its functionality as a fission product barrier (Gerczak et al., 2020). A deterministic performance assessment for deep-burn modular high-temperature reactors for disposal in the proposed repository at Yucca Mountain suggested that the lifetime of the graphite matrix exceeds that of the TRISO particles (van den Akker and Ahn, 2013). This result depends very much on the redox conditions, and TRISO kernels have a low durability in the highly oxidizing conditions of the Yucca Mountain repository. The analysis does not explicitly capture the variations in chemical behavior as a function of chemical conditions (e.g., congruent release of radionuclides with graphite matrix corrosion); however, despite these uncertainties, the results indicate that the release of radioactivity complies with federal dose standards.

Processing and Treatment of TRISO Fuel Elements

As mentioned in Section 5.4.2 and Box 5.3 in Chapter 5, graphite-moderated reactors will generate considerable volumes of radioactive graphite, and as a result, the direct disposal of used prismatic fuel blocks (or graphite pebbles of pebble-bed modular reactors [PBMRs]) is not considered viable (Grambow et al., 2006). Techniques such as low-temperature ($H_2SO_4 + H_2O_2$) and high-temperature ($H_2SO_4 + HNO_3$) acid treatments can be used to separate kernel fuel coated particles from the graphite matrix with high efficiency and nearly complete separation (Guittonneau et al., 2010). Processing TRISO fuel assemblies and pebbles may allow for the embedding of TRISO fuel kernels into waste form matrices (e.g., nuclear waste glass, SiC), which would substantially reduce the volume of disposed waste (Grambow et al., 2006). The vitrification of fuel kernels can be accomplished by glass melting (1200–1300°C) or sintering (700°C). The lower temperature of sintering has the attractive potential of conserving as much as possible the strong confinement properties of the fuel kernels. These lower-temperature processing techniques are under investigation but have already shown promising results. Embedding the kernels in SiC using standard techniques of liquid sintering or hot pressing (1800–2000°C), on the other hand, may damage fuel particles and cause the release of fission products.

Future Research Needs for TRISO Fuel

Despite the significant research efforts on TRISO fuel properties described above, additional research is required to better understand the behavior of TRISO fuel under irradiation and in a repository environment, as well as to improve methods for processing spent TRISO fuel and managing radioactive graphite wastes.

The coatings on the kernel particles experience radiation damage from in-reactor neutron irradiation, fission fragments, and alpha decay events (i.e., alpha particle and recoil nucleus), and the different radiation effects must be understood over the appropriate timescales and temperature regimes (Fachinger et al., 2006). In-reactor formation and diffusion of fission product elements and uranium has been shown to lead to their accumulation at the interfaces of the layers and to some degradation of the layers (Gerczak et al., 2020). Developing a broader understanding of the behavior of radionuclides in TRISO fuels during reactor operation and over the long periods required for their isolation in a geologic repository would be valuable, in particular understanding the effect of irradiation as a function of temperature and radionuclide formation, diffusion, and release through the SiC layer (Gerczak et al., 2020). An additional aspect to understand is "the influence of the internal pressure within the coated particles on the lifetime of the coating and new phase formation between the fuel and the coating layers" (Fachinger et al., 2006). SiC has been selected as a coating-material because of its neutron radiation damage tolerance at high doses, strength, and chemical inertness, but radiation damage can lead to changes in microstructure that affect the integrity and barrier function of SiC (van Rooyer et al., 2018). The presence of defects will affect fission product transport through the SiC layer, so experimental data (e.g., by ion beam or in-reactor irradiations) on the formation, types, and migration of radiation-induced defects is needed (van Rooyer et al., 2018). Additionally, the dissolution behavior of fuel kernels, "especially with respect to the influence of irradiation and incorporation of fission products," and the potential for "rapid release from segregated, radionuclide phases" must be understood (Fachinger et al., 2006).

For disposal of spent TRISO fuel, the safety assessment will require integrated models that describe the release of radionuclides over periods relevant to a safety analysis of a geologic repository. The performance of TRISO fuels must be investigated as a function of external parameters: type and dose of radiation; type of aqueous phase–solid interactions as a function of solution composition and temperature (Fachinger et al., 2006). Because the graphite matrix provides a barrier to the contact of groundwater with the TRISO fuel kernels, it is important to investigate the mechanisms by which water may penetrate the graphite matrix and understand the effect of aqueous phases on the microstructure of the graphite matrix (Fachinger et al., 2006). The release of radionuclides may, in some cases, be limited by radionuclide solubility constraints. Understanding the role of solubility limits on radionuclide mass transfer in the graphite pore fluids should be investigated within the constraints of hydrodynamic movement of fluids and the pore fluid chemistry, and considering that these phenomena will be affected by the radiolysis of water and the corrosion of graphite (Fachinger et al., 2006). For TRISO fuel kernels embedded in a nuclear waste form matrix (e.g., borosilicate glass, SiC, or ZrO_2), outstanding questions include the following: (1) Are sintered matrices

(glass and ZrO_2) sufficiently dense to prevent water diffusion toward the fuel particles without the dissolution of the matrix? (2) Once the embedding matrix has altered to secondary phases, which will require hundreds of thousands of years, will the matrix still protect the fuel particles? (3) What are experimentally determined values of the diffusive mass transfer of radionuclides across a corroded embedding matrix? (Grambow et al., 2008)

Detailed investigations of the corrosion mechanisms of irradiated pyrocarbon and SiC layers are also still required (Fukuda et al., 1982; Grambow et al., 2010). There have been proposals for the importance of the graphite matrix in reducing the release of radionuclides from TRISO fuels; however, before one can claim credit for this barrier function, there is a need for detailed materials studies of the mechanisms of graphite corrosion in repository environments (van den Akker and Ahn, 2013). Similarly, although degradation measurements of carbonaceous materials relevant to TRISO fuel behavior in a geologic repository are available, the data are sparse. Confirmatory experiments need to be completed over the appropriate temperature and pH ranges for the appropriate solution compositions that are expected in different geologic settings (Morris and Bauer, 2005). "A better understanding of the corrosion rates of the O[outer]PyC and SiC layers, along with increasing the quality control of the OPyC and SiC layer thicknesses, can significantly reduce the uncertainty in estimated time to failure of spent TRISO fuel in a repository environment" (Peterson and Dunzik-Gougar, 2011). For long-term, safe disposal of kernels from TRISO fuel, it is important to investigate (1) the effect of radioactive alpha decay of actinides on the structural integrity of the various coatings, including the atomic scale radiation damage effects and volume expansion; (2) He accumulation in the particle kernels that may lead to pressurization and increase the probability of particle coating failure; (3) the effects of lithostatic pressure; and (4) material fatigue mechanisms (Nabielek et al., 2009). It may be possible to improve on the present TRISO particle design, for instance by including a thin layer of ZrC among the coatings to obtain better performance at higher temperatures (Malherbe, 2013).

The management of irradiated graphite waste represents a considerable research and development challenge regarding retrieval, characterization, treatment, reuse, and final disposal. The activation of impurities, which vary considerably depending on the grade of the graphite, during neutron irradiation of the graphite determines the degree of radionuclide contamination. However, the European Commission CARBOWASTE project has demonstrated that a number of the radionuclides (e.g., ^{3}H, ^{36}Cl, ^{60}Co, and ^{63}Ni, as well as ^{14}C) can be removed with high efficiency (von Lensa et al., 2011). Previous research on TRISO processing has demonstrated that kernels can be separated from the graphite matrix (Guittonneau et al., 2010), but this work must be expanded to study the effects of neutron irradiation on the compacts used within the reactor and to understand the fate of volatile elements, such as ^{36}Cl and ^{14}C, during the processing treatment.

Appendix H

Reprocessing and Recycling Practices in Other Countries

Most of the approximately 30 countries with nuclear power programs neither reprocess spent nuclear fuel nor use mixed oxide (MOX) fuels. Only France and Russia currently operate commercial-scale reprocessing facilities. China has one operating small-scale facility for reprocessing civilian nuclear fuel and one under construction. About 10 percent of the world's reactors are licensed to use MOX fuel, but MOX comprises only about 5 percent of the world's new nuclear fuel (WNA, 2017a).

In 2010, the Electric Power Research Institute (EPRI) conducted an extensive review of advanced fuel cycle strategies looking at sustainability related to four challenges: (1) natural uranium resources, (2) waste management, (3) economic competitiveness, and (4) nonproliferation (EPRI, 2010b). That review evaluated closing the fuel cycle first in terms of Pu management, which has a significant impact on storage decisions, and then in terms of management of the minor actinides (Np, Am, Cm) by considering a sequence of advanced fuel cycles that are described in Chapter 4. EPRI concluded that the real strategic choice, from a technical perspective, is either the once-through fuel cycle (i.e., no plutonium recovery) or the reuse of the recovered Pu in fast reactors for multirecycling. Only those countries that commit to pursuing the latter option will potentially benefit from their involvement in developing the industrial know-how necessary to adapt the PUREX process to the reprocessing of light water reactor (LWR) oxide fuels. Otherwise, they would later be faced with introducing even more advanced reprocessing technologies, such as reprocessing of fast reactor fuels, without extensive prior experience (EPRI, 2010b).

Below are summaries of reprocessing activities in China, India, Japan, Russia, and the United Kingdom. Section 2.6 provides more detailed descriptions and a comparison of the relevant programs and policies in France and the United States.

CHINA

China's overarching goal in the development of its nuclear power program is to become self-sufficient in most aspects of the nuclear fuel cycle. Dating back to the 1980s, China opted for a closed fuel cycle strategy, motivated by desires to (1) increase uranium resource utilization and reduce costs associated with mining, milling, and enrichment; (2) increase energy security; (3) decrease repository volume; (4) minimize waste radiotoxicity; and (5) reduce spent fuel in reactor pools (Zhang, 2021). One of the strategic principles for nuclear power in China's 13th Five Year Plan, which was approved in March 2016, was to accelerate the building of demonstration

and large-scale reprocessing plants. In 2010, China had completed commissioning of a pilot reprocessing plant at Lanzhou Nuclear Fuel Complex and used it to reprocess about 50 MT (metric tons) of spent fuel from 2013 to 2015. China next began work on a demonstration spent fuel reprocessing plant with a capacity of 200 MT annually, which is being constructed in Gansu Nuclear Technology Industrial Park and is expected to start operations in 2025 (WNA, 2021h). A MOX demonstration plant with 20 MT per year capacity is being built at the same site (Zhang, 2021). In addition, since the end of 2020 China has reportedly been constructing a second 200-MT/y plant (WNA, 2021h).

To reach commercial-scale operations of reprocessing, China has partnered with French-based Areva (now Orano) since November 2007 to develop an 800-MT/y reprocessing and MOX fabrication plant using French technology (WNA, 2021h). In August 2021, Orano announced that negotiations were in the final phase and that China's plant will be based on Orano's La Hague reprocessing plant and the MELOX MOX fabrication plant (Orano, 2021b). The Chinese National Nuclear Corporation aimed to begin construction of this 800-MT/y plant in 2020 and start operation by 2030, but as of 2021 no site had been selected (Zhang, 2021).

INDIA

Since the 1950s, India has had a three-part strategy for its nuclear power program: (1) using limited indigenous supplies of natural uranium to fuel a fleet of heavy water reactors on natural uranium fuel, (2) reprocessing this spent fuel to recycle plutonium to fuel breeder reactors, and (3) using the produced plutonium in a Th fuel cycle to breed ^{233}U (see Chapter 3). India has about one-third of the known global Th supplies.

While India has successfully executed the first phase of its strategic plan, it has not fully developed the other two phases. India has a modest total reprocessing capacity of about 200 MT/y with about 100 MT annual capacity at each plant at Tarapur and Kalpakkam. About 115 MT/y are reprocessed, resulting in about 400 kg of Pu annually; these reprocessing plants could be used for weapons purposes. Under the 2010 U.S.–India nuclear deal, India announced plans to build two larger capacity reprocessing plants that would be placed under International Atomic Energy Agency (IAEA) safeguards (WNA, 2021g).

JAPAN

Japan has pursued development of reprocessing and Pu recycling since the beginning of its nuclear energy program in 1956. From 1977 to 2009, at Tokai, the Japan Atomic Energy Agency operated a pilot-scale reprocessing plant with 90 MT annual capacity. During more than three decades of operations until its closure in 2014, the Tokai plant reprocessed 1,140 MT of spent fuel, including spent fuel from its fleet of boiling water reactors, pressurized water reactors, and Pu-U MOX and reprocessed U fuel from its heavy water–moderated, light water–cooled Advanced Thermal Reactor (JAEA, n.d.; WNA, 2021f).

For more than three decades, Japan has been building the Rokkasho Reprocessing Plant (RRP), which has a planned capacity of 800 MT/y. The startup of RRP has been long delayed in part because Japan had to rebuild the vitrification plant and retrofit additional safety and security features throughout the plant. As of August 2020, the total project cost estimate was ¥13,900 billion ($130 billion) to include construction and operation of the RRP and its eventual decommissioning (NEI, 2020b). In February 2022, Japan Nuclear Fuel Limited (JNFL), which will operate RRP, stated that actual reprocessing operations would likely not occur until fiscal year (FY) 2023, in part because it had to meet new regulatory requirements for safety upgrades imposed after Fukushima. JNFL also announced a delay in the startup of the MOX fuel fabrication plant to the first half of Japan's FY2024 (JNFL, 2022). As with the RRP, JNFL will have to install additional safety measures at the MOX plant and will have to undergo additional regulatory agency approval of this work before it will be allowed to operate.

In the aftermath of the 2011 Fukushima accident, the Japanese government reevaluated its nuclear energy policy and decided to substantially reduce the share of nuclear power in electricity generation from the 30 percent level at the time of the accident to a projected 20–22 percent by 2030. (In early April 2022, amid the crisis in Ukraine, Prime Minister Fumio Kishida floated the idea that Japan might increase its use of nuclear power.) While the post-Fukushima accident policy shift might have also resulted in Japan's reducing or even abandoning its

planned use of reprocessing, the Japanese government is still committed to reprocessing 100 percent of its spent fuel because Japanese utilities view Pu as an asset. If the government would abandon reprocessing, the utilities' balance sheets would then have to book spent fuel as a liability, potentially tipping some utilities into bankruptcy unless the government bails them out. Another reason for the continued reprocessing policy is the government's commitment to local communities around nuclear power plants and the reprocessing plant that spent fuel would not end up indefinitely stored on the sites and that Japan would eventually close the fuel cycle, although that possibility has been delayed beyond 2050 (Toki and Pomper, 2013). Most of Japan's reprocessed spent fuel was done in other countries, namely France and the United Kingdom, which have accumulated a large stockpile of Japanese origin plutonium. Japan's Spent Nuclear Fuel Reprocessing Fund Act, passed in 2016, outlines a plan for funding and implementing the reprocessing of spent fuel (METI, 2016).

RUSSIA

Russia considers Pu a strategic energy resource, and its policy is to recycle as much reprocessed U and Pu as possible. However, the practical application has fallen short, with only about 16 percent of used fuel having been reprocessed. Still, Russia's stated goal is to fully close the fuel cycle by 2030 (WNA, 2021d).

Russia has performed commercial reprocessing since the 1970s, with operation of the RT-1 plant at the Mayak Chemical Combine in Ozersk beginning in 1971. RT-1 has a capacity of 400 MT/y, but it has not been operating at full capacity in recent years because of environmental constraints and the loss of foreign reprocessing contracts. The much larger-capacity RT-2 plant at the Mayak Chemical Combine in Zheleznogorsk is projected to be ready by 2025 and has a planned capacity of 700 MT/y. An additional 800 MT of annual capacity are slated to come online a few years after the RT-2 start date (WNA, 2021d). Also, Zheleznogorsk houses the Pilot Demonstration Center for researching and demonstrating innovative reprocessing methods for both fast and thermal reactors. (NEI, 2017).

Russia has significant experience in commercially operating fast neutron sodium-cooled reactors. The BN-350 (350 MWe) operated in Aktau, Kazakhstan, from 1973 to 1999, after which it was decommissioned. The BN-600 (600 MWe) has been in operation since 1981 in Zarechnyy at the Beloyarsk Nuclear Power Plant, and its license is to be extended to 2040. These reactors have been fueled with 17, 21, and 26 percent–enriched uranium. The BN-800 (800 MWe), also at Beloyarsk, has been operational since its connection to the grid in December 2015. Because of limits on MOX fabrication capacity, the BN-800 started up with a hybrid core of highly enriched uranium plus experimental MOX; a core partially fueled with standard pelletized MOX has been used since August 2019, and a transition to a full MOX core occurred in 2022. Uranium-plutonium nitride fuel is under development. At Seversk, Russia is moving forward on the "breakthrough project," an experimental demonstration power complex designed to demonstrate a closed fuel cycle. The project consists of a power reactor in combination with a reprocessing plant and fuel fabrication facility. The reactor is the Brest-OD-300, a fast-neutron, lead-cooled reactor rated at 300 MWe, 700 MWth, that will be fueled by a dense mixed uranium-plutonium nitride fuel. Construction began in June 2021 and is expected to be completed in 2026–2027. Russia's other advanced reactor projects, which could use recycled fuel, include (1) the BN-1200, which would be a fast neutron sodium-cooled reactor at 1200 MWe using nitride fuel (the decision to build was postponed until 2030); (2) the SVBR-100, which would be a fast neutron lead-bismuth-cooled reactor based on naval reactor design; and (3) molten salt reactors, which are undergoing research and development (Podvig, 2021).

Russia is also developing and promoting a new option for recycling spent nuclear fuel in thermal reactors using REMIX fuel (REgenerated MIXture of U-Pu oxides), which involves mixing together reprocessed U and Pu and increasing the ^{235}U content to the required level by the addition of HALEU (<20 percent ^{235}U). The resulting neutron spectrum of REMIX fuel does not differ significantly from standard enriched UOX, and the Pu content is nominally less than 2 percent. As a result, REMIX fuel can be used without reactor modifications at 100 percent core loading and can be recycled around 5 times in an LWR. Russia claims that the REMIX concept will allow more efficient use of nuclear fuel, reduce the amount of spent nuclear fuel to be stored and disposed, and decrease the risk of nuclear proliferation (Postovarova et al., 2016). In December 2021, Russia announced that six REMIX

fuel assemblies were loaded into a VVER-1000 reactor (Balakovo NPP) for a planned 5-year irradiation period (NEI, 2021b).

UNITED KINGDOM

In the United Kingdom, reprocessing efforts were motivated by the necessity to stabilize spent Magnox fuel prior to storage and for the production of Pu for nuclear weapons. Magnox reactors, which have all been shut down, ran on natural uranium metal fuel and were graphite-moderated and gas-cooled. The United Kingdom reprocessed all Magnox spent fuel 6 months after removing it from a reactor because corrosion of the cladding does not allow the spent fuel to be stored for long periods underwater (Worrall, 2021). The B205 Magnox Reprocessing Plant, which began operations in 1964 at Sellafield, was scheduled to close at the end of 2021 after reprocessing the final Magnox spent fuel (IPFM, 2020), but was still operating as of February 2022 (ENS, 2022).

The United Kingdom has also reprocessed oxide fuels. In 1994, the United Kingdom commissioned the Thermal Oxide Reprocessing Plant (THORP) with a capacity of 600 MT/y. In November 2018, THORP was closed after having processed more than 9,000 MT of spent fuel from 30 customers in nine countries after a loss of overseas contracts made continued operation economically infeasible. UK nuclear power plants provided about 60 percent of the spent fuel that THORP reprocessed. The THORP storage pool is being used to store some Advanced Gas Reactor spent fuel and other types of UK reactors' spent fuel while awaiting final disposal (WNA, 2021e).

The United Kingdom has a backlog of about 140 MT of stored Pu (24 MT of which are foreign owned) (IAEA, 2021c). UK utilities are not interested in reusing the Pu in their nuclear power plants because U-based fuels are less expensive and U supplies are relatively abundant. However, since 2011 the UK government policy is to demonstrate the feasibility of MOX fuel and to use it in commercial reactors. (The UK fast reactor program was shut down in 1994.) However, ^{241}Am has built up in the stored plutonium due to the decay of ^{241}Pu and will have to be removed to make MOX fuel. Research is ongoing to demonstrate that "the blending of plutonium batches will enable the americium-241 ingrowth to be adequately managed with respect to MOX fuel manufacture" (Hyatt, 2020).

Appendix I

Statement of Task for Parallel National Academies' Study Laying the Foundation for New and Advanced Nuclear Reactors in the United States

The National Academies of Sciences, Engineering, and Medicine will appoint an ad hoc committee of experts to identify opportunities and barriers to the commercialization of new and advanced nuclear reactor technologies in the United States over the next 30 years as part of a decarbonization strategy. Specific topics the committee will examine include:

- The research, development, and demonstration needed for new and advanced nuclear reactor technologies to reach commercial readiness, the potential for leveraging technological developments outside the nuclear energy sector, and the manufacturing, construction, financial, societal, and other barriers associated with their deployment;
- The operational characteristics of these technologies, including their implications for safety, security, and nonproliferation, as well as their interaction with other low-carbon generation and storage resources that may be relevant to a changing electricity system;
- The economic, regulatory, and business challenges associated with commercialization of these technologies;
- The implications of these technologies for the front and back end of the fuel cycle;
- The viability of these technologies in applications outside the electricity sector, for example in desalination, water and wastewater treatment, hydrogen production, or process heat;
- The role of the U.S. Government in sponsoring the development and commercialization of new and advanced nuclear reactor technologies to provide clean energy, to address national-security and nonproliferation goals, or to assist in nuclear exports; and
- The future workforce and educational needs to support the research, development, and deployment of these technologies.