APPENDIX A

BACKGROUND INFORMATION ON NUCLEAR FUEL CYCLE TECHNOLOGIES AND THE ADVANCED FUEL CYCLE INITIATIVE

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This appendix provides background information related to the domestic programmatic alternatives, including the Advanced Fuel Cycle Initiative (AFCI) (which would continue under any alternative), analyzed in this Global Nuclear Energy Partnership (GNEP) Programmatic Environmental Impact Statement (PEIS). The information in this appendix fulfills two primary objectives. First, this PEIS involves a technically complex subject matter. To make subjects such as nuclear fuel cycles, spent nuclear fuel (SNF), reprocessing, and transmutation of long-lived radiotoxic materials more easily understood, this appendix presents a background discussion of nuclear power concepts, technologies, and terminology.

Second, the analysis of potential environmental impacts associated with the domestic programmatic alternatives (contained primarily in Chapter 4) is based on broad implementation of several technologies. In general, that analysis is based on deployment of multiples of a single facility, or variations on the size of a facility. This appendix describes the reference facilities used as a basis for the programmatic analysis. For example, this appendix provides details about a typical light water reactor (LWR) (the type currently used for nuclear power in the United States). The programmatic analysis in Chapter 4 describes impacts associated with the deployment of about 100 to 400 such reactors.

The alternatives in this PEIS involve technologies that vary significantly in their state of development and commercial use. As a result, the amount of information available about these technologies also varies. Relatively less information is available regarding these technologies which are essential components of certain domestic programmatic alternatives: fast reactors, deep burn high temperature gas-cooled reactors (HTGRs), nuclear fuel recycling facilities, and fuel fabrication technologies for transmutation fuel and thorium-fuel. Where information is incomplete and unavailable, DOE developed preliminary design information and/or discussed technologies by comparison to other technologies for which sufficient information is available.

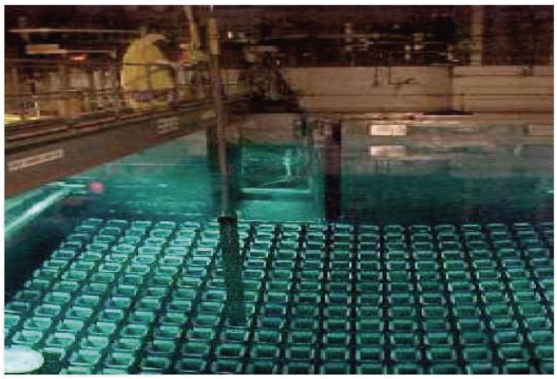
A.1 NUCLEAR FUEL CYCLE

What does the term “nuclear fuel cycle” mean?

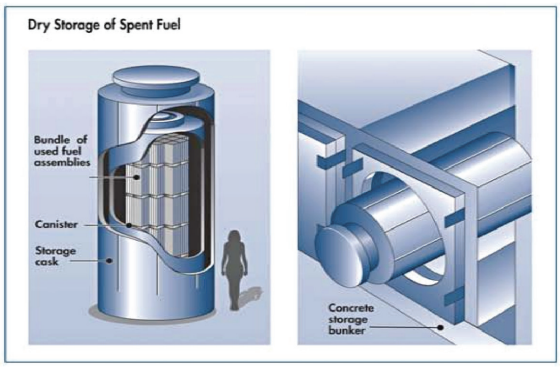
A “fuel cycle” refers to the life cycle of nuclear fuel, including the sources of fuel, its application, and disposition. The cycle consists of "front end" steps leading to the preparation of fuel for reactor operation and "back end" steps leading to the safe management and disposal of the highly radioactive SNF. The fuel cycle starts with mining and refining uranium ore. Generally, uranium (U) must be “enriched” (to approximately 3 to 5 percent) in the isotope U-235 which fissions more easily than other natural uranium isotopes. The enriched

uranium is then made into fuel for use in a commercial nuclear power plant. This appendix discusses uranium mining, refining, enrichment, and the manufacture of nuclear fuel.

After an operating cycle (typically 18 months), the reactor is shut down for refueling. Although the SNF that comes out of a nuclear reactor is only partially consumed, it is highly radioactive and must be stored in specially designed pools, resembling large swimming pools, which provide both cooling (the SNF continues to generate heat as a result of residual radioactive decay) and shielding (to protect the environment from residual ionizing radiation) (see Figure A.1-1). After sufficient time has passed to reduce the SNF's thermal output (approximately 3 to 5 years), it may be desirable to store SNF in dry storage facilities (see Figure A.1-2). Eventually, the SNF would be disposed of in a geologic repository or recycled.

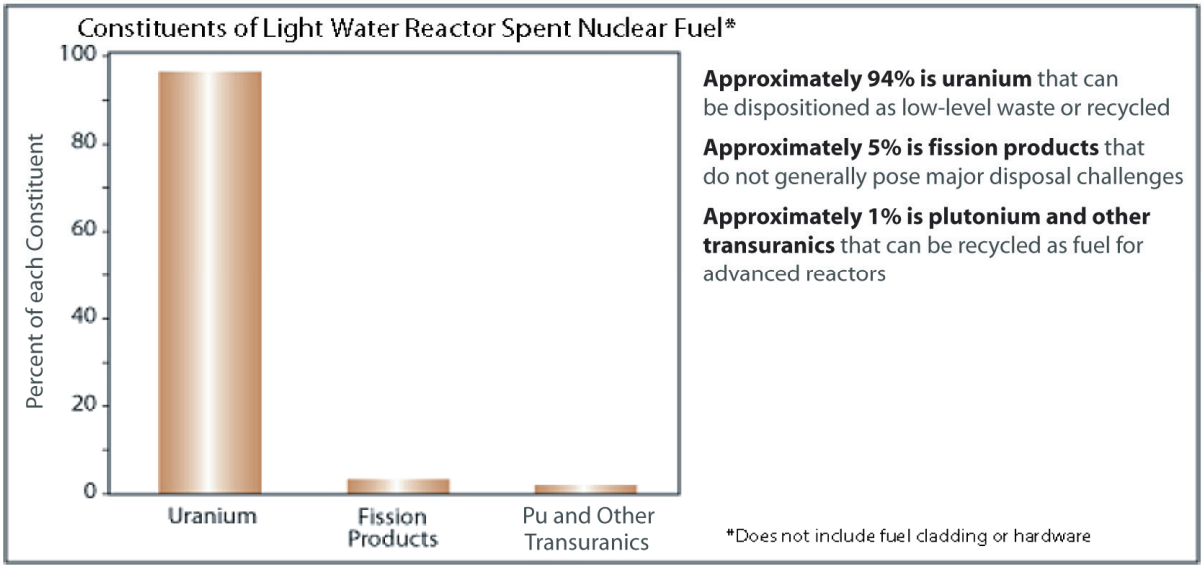


Source: Holtec 2008
FIGURE A.1-1—*SNF Assemblies Being Loaded Into a Typical SNF Pool*



Source: NRC 2008b
FIGURE A.1-2—*Typical Dry Cask Storage*

As shown on Figure A.1-3, SNF from a typical commercial LWR fuel has three major components: uranium (96 percent), fission products (3 percent) such as cesium (Cs) and strontium (Sr), and transuranic (TRU) elements (1 percent) such as plutonium (Pu). Because SNF is approximately 96 percent uranium, removing this material for reuse or disposal as low-level waste (LLW) would result in a reduction in the volume of waste requiring geologic disposal. Compared to the transuranic elements, most of the fission products are relatively stable and short-lived and do not pose major disposal challenges if separated from the SNF. The transuranic elements contain significant amounts of stored energy and include a number of the longest-lived radiotoxic isotopes, some of which can be used in a nuclear weapon. Ultimately, the two strategies for managing SNF are to dispose of the used fuel as waste, or to separate and recycle the uranium and transuranic elements from the waste products. These two divergent strategies for managing SNF lead to two very different nuclear fuel cycles: "open cycle" and "closed cycle."

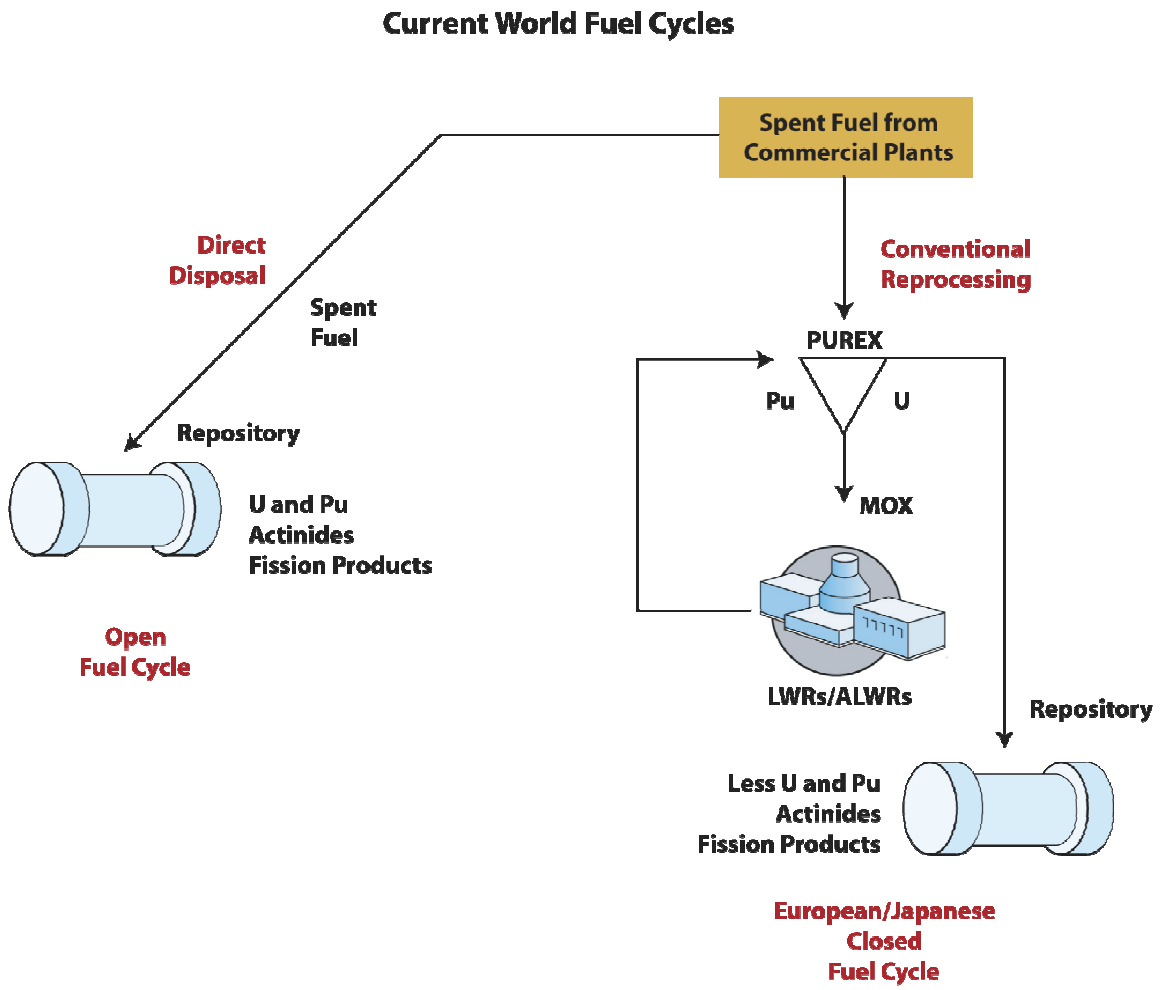


Source: Derived from DOE 2006u

FIGURE A.1-3—*Constituents of Light Water Reactor Spent Nuclear Fuel*

What is the difference between an "open" and "closed" fuel cycle?

Figure A.1-4 depicts both an open and closed nuclear fuel cycle. In the open fuel cycle (also known as the "once-through fuel cycle" and/or "direct disposal"), reactor fuel is used in a nuclear power plant only once. Although this approach to managing SNF accumulates liabilities such as transuranics, this is the simplest fuel cycle as no separation plants are needed and no fabrication of recycled material is needed. The United States and many other countries presently use this approach.



Source: INL 2003

FIGURE A.1-4—*Open and Closed Nuclear Fuel Cycle*

SNF discharged from LWRs contains appreciable quantities of fissile material such as U-235 and Pu-239, as well as other radioactive materials. These fissile materials can be separated and recovered from the SNF. In the closed fuel cycle, the SNF is recycled and some of the usable constituents are made into new reactor fuel (note: Figure A.1-4 identifies the Plutonium and Uranium Reduction and Extraction (PUREX) process for reprocessing SNF and producing mixed-oxide (MOX) fuel; however, there are several other reprocessing technologies that can be used to separate SNF and produce various types of reactor fuels with transuranic elements). By removing the uranium, fission products, and transuranics, recycling SNF can reduce the volume, radiotoxicity, and heat load of waste requiring geologic disposal.

As discussed in the next section, "transmutation" is the primary method of destroying the transuranic elements that are removed. In this process, the transuranic elements are consumed and converted into fission products and additional energy is produced. The closed fuel cycle extracts energy from the transuranic elements, thereby making them energy assets instead of waste liabilities. Countries such as France, the United Kingdom, Russia, and Japan use this approach.

A.2 TRANSURANIC ELEMENTS

Much of the discussion in this PEIS involves **transuranic elements** and the potential approaches to managing the transuranic elements that are contained in SNF. Transuranic elements are created in nuclear reactors when uranium in the fuel absorbs or captures neutrons. Transuranic elements constitute approximately 1 percent of SNF. Certain transuranic elements can be used in nuclear weapons. The transuranic elements are generally long-lived (tens of thousands to millions of years) and remain significantly more radiotoxic than the original uranium ore for hundreds of thousands of years. Certain transuranics also are significant contributors to the long-term thermal output of SNF. Additionally, transuranic elements and enriched uranium are the primary materials in SNF of concern for nuclear weapons proliferation.[1]

Transuranic Elements

All matter is divided into about 100 different chemical elements. The lightest element is hydrogen, #1. Element #92 is uranium, which is the heaviest element found in nature in significant amounts. Transuranic elements are man-made elements that are heavier than uranium, and include neptunium, plutonium, americium, and curium. Transuranic elements are created in nuclear power plants when uranium absorbs or captures neutrons. Transuranic elements are generally long-lived and radiotoxic, and certain transuranic elements can be used in nuclear weapons.

If the transuranics can be destroyed—through a process known as transmutation—then the waste hazard would be dominated by fission products. Fission products generally decay much more quickly (over a few hundred years for the more common fission products), thereby reducing the long-term hazard relative to transuranic elements. Transmutation is the conversion of one isotope to another by changing its structure, which can change both its nuclear and chemical properties. Transmutation can occur within a nuclear reactor, while generating electricity.

Why Transuranic Elements Are Important

- Transuranics have a major effect on long-term heat load and long-term radiotoxicity of SNF.
- Transuranics and enriched uranium are the primary materials in SNF of concern for nuclear weapons proliferation.
- Transuranics can be destroyed while producing energy if recycled in reactors.

A.3 MINING, ENRICHMENT, AND FUEL FABRICATION

A.3.1 Uranium

Uranium is a naturally-occurring element found at low levels in virtually all rock, soil, and water. Significant concentrations of uranium occur in some substances such as phosphate rock deposits, and minerals such as uraninite in uranium-rich ores. Because uranium has such a long radioactive half-life (4.5 billion years for U-238), the total amount of it on earth stays almost the same. When refined, uranium is a silvery white, weakly radioactive metal. Uranium metal has a

[1] By itself, low enriched uranium (LEU) is not usable for making a nuclear weapon; however, LEU in SNF could be a proliferation concern if used in a radiation dispersal device. LEU used to fuel reactors produces plutonium, which, if separated from SNF, can be used to make a nuclear weapon. Additionally, it is possible to use LEU as a feed material that could be enriched to high enough levels to make a nuclear weapon.

very high density (65 percent more dense than lead). Uranium in ores can be extracted and chemically converted into uranium dioxide or other chemical forms usable in industry (EPA 2005b).

A.3.1.1 *Uranium Mining*

Uranium ore is mined either by the open pit method (to recover deposits near the surface of the earth) or by underground mining. Once recovered, the ore is crushed and then ground. Uranium ore rich in vanadium is usually roasted with sodium chloride (salt) or soda ash prior to grinding in order to facilitate extraction of the ore with water (EPA 2005a). Two methods are employed to extract uranium from ore: 1) acid leaching with sulfuric acid, or 2) alkaline leaching with a hot solution of sodium carbonate and sodium bicarbonate. This leaching can occur *in situ*, which involves injecting solutions into the permeable ore zone, or as *solvent extraction*, where the ore is placed in tanks and the same solutions are introduced. Ion exchange can also be used for both the in situ and solvent extraction methods, utilizing fixed organic resins (EPA 2005a).

A.3.1.2 *Uranium Milling and Extraction*

Concentrated uraniferous (uranium containing) ions from solvent extraction or ion exchange units are precipitated out of solution to produce a material referred to as "yellowcake." Uranium is usually precipitated from acid solutions by neutralization with ammonia or magnesia. Hydrogen peroxide may also be added to an acid pregnant stripping liquor or pregnant elution liquor to precipitate uranium peroxide (EPA 2005a). Additional chemical procedures fluoridate the yellowcake and then remove the fluoride to render relatively pure, uranium metal. If the uranium is to be enriched for purposes such as use as reactor fuel, the fluoride is not removed and the intermediary product uranium hexafluoride is used directly as a feedstock for the enrichment process.

A.3.1.3 *Uranium Enrichment*

For most reactor types, uranium needs to be enriched from the yellowcake state in order to produce material sufficiently concentrated to sustain a fission process which can generate the desired heat factor. Natural uranium primarily contains two isotopes, U-238 (99.3 percent) and U-235 (0.7 percent) (NRC 2007h). The concentration of U-235, the fissionable isotope in uranium, needs to be increased to 3 to 5 percent for practical use as a nuclear fuel in a LWR. Gaseous diffusion and gas centrifuges are the principal methods for enrichment (NRC 2007h).

In gaseous diffusion, uranium is converted into a gaseous form, uranium hexafluoride (UF_6), and passed through many stages of barriers that separate the uranium isotopes. In the United States, gaseous diffusion plants have operated in Oak Ridge, TN; Paducah, KY; and Piketon, OH. Currently, the only operating enrichment plant in the United States is the plant in Paducah (NRC 2007h).

Another way to enrich uranium is by using gas centrifuges. Gas centrifuges spin UF_6 gas at high speeds creating a centrifugal force that separates the isotopes by forcing the heavier U-238 further outward in the centrifuge. Gas centrifuges have been used in Europe for about 30 years

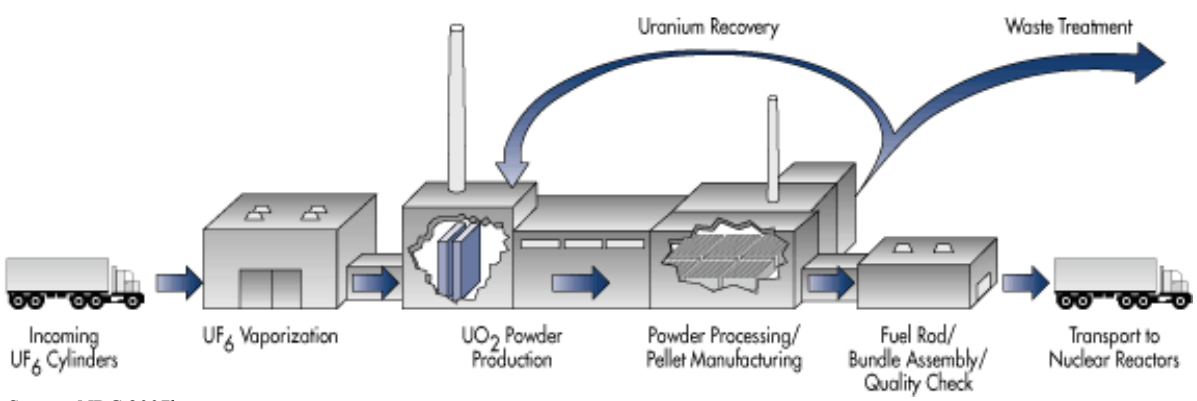
for enriching uranium (NRC 2007h) and two such facilities are under construction in the United States—a Louisiana Energy Services facility in Eunice, NM, and a U.S. Enrichment Corporation facility in Piketon, OH. Gas centrifuge is currently preferred, in part, because it requires less electricity than gaseous diffusion.

Enrichment processes for LWRs generate a product from 3 to 5 percent U-235 for use as nuclear fuel and a byproduct of depleted uranium (about 0.3 percent U-235). Depleted uranium has some commercial applications in counterweights, antitank armaments, and ammunition; however, the commercial demand for depleted uranium is much less than the amounts generated (NRC 2007h). There are also hazardous and mixed wastes generated as a result of the enrichment process. These wastes must be managed in accordance with all applicable requirements.

A.3.1.4 *Uranium Fuel Fabrication*

This section discusses uranium fuel fabrication for the various fuel types that could be needed for the domestic programmatic alternatives. This section first discusses uranium oxide fuel fabrication for LWRs, which is the predominant fuel fabrication process used in the United States. This section also discusses uranium oxide fuel fabrication for heavy water reactors (HWRs) and high temperature gas-cooled reactors (HTGRs); mixed-oxide (MOX) fuel fabrication; and transmutation fuel fabrication. Thorium fuels are discussed in Section A.3.2.

Light Water Reactor Uranium Oxide Fuel. Fuel fabrication for LWRs typically begins with the receipt of UF_6 in solid form from an enrichment plant. The UF_6 is heated to a gaseous form, and the UF_6 gas is chemically processed to form uranium dioxide (UO_2) powder. This powder is then pressed into pellets, sintered into ceramic form, loaded into zircaloy tubes, and constructed into fuel assemblies. Depending on the type of LWR, a fuel assembly may contain up to 264 fuel rods and have dimensions of 5 to 9 square inches (in^2) (32 to 58 square centimeters [cm^2]) by about 12 feet (ft) (3.6 meters [m]) long (NRC 2007c). Figure A.3.1.4-1 shows a typical LWR fuel fabrication facility.



Source: NRC 2007h

FIGURE A.3.1.4-1—*Typical Nuclear Reactor Uranium Dioxide Fuel Fabrication Facility*

Heavy Water Reactor Uranium Oxide Fuel. While LWRs utilize uranium in the form of UF_6, so that it can be enriched to 3 to 5 percent, HWRs, such as the Canada Deuterium Uranium (CANDU) and Indian reactors, use UO_2 which is normally not enriched. Current CANDU

designs do not need enriched uranium to achieve criticality (due to their more efficient heavy water moderator), however, some newer concepts call for low enrichment to help reduce the size of the reactors (WNA 2008d). To make the fuel pellets for a HWR reactor, the uranium dioxide is compressed, then baked at high temperatures to yield hard, insoluble, ceramic cylinders about 0.6 inch (in) (14 millimeter [mm]) in diameter by 0.8 in (20 mm) long. These uranium dioxide pellets , which are about the same size as the UF_6 pellets used in LWR fuel, are then stacked about 20 in (500 mm) long and 4 in (100 mm) in diameter and encapsulated in thin walled tubes of zirconium alloy (Canada 2008). Uranium dioxide is a very concentrated energy source. A number of such fuel tubes, usually referred to as pins, are assembled to form a fuel bundle that can be conveniently loaded into the reactor. Each bundle is roughly 44 pounds (lbs) (20 kilograms [kg]), and a typical core loading is on the order of 4500 bundles. See Section A.5.3 for a discussion of HWR fuel fabrication associated with the Thermal Recycle Alternative (Option 2).

High Temperature Gas-Cooled Reactor Uranium Oxide Fuel. In many respects, the fuel fabrication process for an HTGR would be similar to the fuel fabrication process described above for LWRs. Enriched uranium would be made into fuel rods, which would be bundled together into fuel assemblies for insertion into a reactor core. A major difference, however, would be the fuel particles that make up the fuel. Typical LWR fuel is made up of uranium powder that has been pressed into pellets, sintered, and loaded into zircaloy tubes. HTGR fuel would be made up of spherical fuel particles (known as kernels) of uranium that are coated with many layers of carbon. These carbon-coated fuel particles are the basic component of the HTGR fuel element (Shropshire and Herring 2004). The carbon layers provide space for gaseous fission products, provide structural strength, act as a containment barrier, and provide a bonding surface. These carbon-coated fuel particles are mixed together in a binder mixture (similar to "tar") to make fuel compacts (Del Cul et al. 2002). The compacts are sintered and inserted into a graphite sleeve to make a fuel rod, and these rods are assembled into graphite blocks to make fuel elements. Currently, only Japan has a fuel fabrication facility to make limited quantities of these carbon-coated fuel particles on a commercial basis.

Light Water Reactor Mixed Oxide Fuel. MOX fuel is fabricated in a similar manner, but differs from low enriched uranium fuel in that the dioxide powder from which the fuel pellets are pressed is a combination of UO_2 and plutonium oxide (PuO_2). MOX fuel fabrication typically begins with blending and milling the plutonium dioxide powder to ensure general consistency in enrichment and isotopic concentration. The uranium and plutonium powders are then blended and milled together to ensure a uniform distribution of the plutonium in the MOX, and to adjust the particle size of the MOX powder. The MOX material, at this point, is placed into fuel rods similar to the LWR uranium fuel rods described above. In the case of a co-separated uranium-plutonium mixture, with or without additional transuranics, a similar approach could be used except that the U-Pu mixture would be blended and milled together with additional uranium in order to obtain the desired mix. Additional shielding might be needed depending on the isotopic composition.

Transmutation Fuel Fabrication. The Fast Recycle Alternative, Thermal/Fast Recycle Alternative, and Thermal Recycle Alternative (Option 3), and possibly Thermal Recycle Alternative (Options 1), would utilize transmutation fuel (fuel made up of transuranic elements

with or without uranium) that would fuel advanced reactors such as the advanced recycling reactor and the deep burn HTGR. The uranium and transuranic products from the separations process would be the feed material for the transmutation fuel fabrication. Given the high radiation fields associated with the transuranic feed material, the fabrication of this fuel must be done remotely, in a shielded facility. Substantial experience exists for fabricating numerous types of reactor fuels, including fast reactor fuel with uranium and plutonium, in both metal and oxide fuel forms, however, these operations did not require the shielding or the remote operations that would apply to the transmutation fuel. The design of the transmutation fuel fabrication processes and facilities do not yet exist but can draw heavily on the past experience with fabricating fast reactor fuel, and would have some unique aspects to address regarding the transuranic fuel and the remote operation.

For a deep-burn HTGR, the transuranic products from the separations process would be the feed material for fuel fabrication. Given the high radiation fields associated with this transuranic product, the fabrication of this fuel must be done remotely, in a shielded facility. Some experience exists for fabricating fast HTGR fuel with uranium and thorium in oxide fuel forms, however, these operations did not require the shielding or the remote operations that would apply to the transuranic fuel. The design of the HTGR transuranic fuel fabrication processes and facilities do not yet exist but can draw on the past experience with fabricating HTGR fuel, and would have some unique aspects to address regarding transuranic fuel and the remote operation. See Section A.6.2 for a description of a transmutation fuel fabrication facility. That section is illustrative of the facility requirements, operations, and waste streams for a transmutation fuel fabrication facility.

A.3.2 Thorium

Thorium is a naturally occurring radioactive substance that can be used in nuclear fuel. Thorium is about three times as abundant as uranium in nature, but cannot, by itself, create or sustain the nuclear chain reaction ("criticality") needed to produce heat to generate electricity. In the environment, thorium exists in combination with other minerals, such as silica. Small amounts of thorium are present in all rocks, soil, water, plants, and animals. Soil contains an average of about 6 parts of thorium per million parts of soil (6 ppm) (ATSDR 1999). Thorium occurs most commonly in the rare earth mineral thorium phosphate (monazite), which contains approximately 12 percent high grade thorium dioxide (ThO_2). There is a large quantity of thorium in the United States and known reserves are extractable in other parts of the world. Table A.3.2-1 shows the world reserves of thorium.

TABLE A.3.2-1—*World Thorium Reserves*

Country	Reserves (Tons)
USA	160,000
Australia	300,000
Brazil	16,000
Canada	100,000
India	290,000
Malaysia	4,500
Norway	170,000
South Africa	35,000
Other Countries	90,000
Total World Deposits	**1,200,000**

Source: USGS 2007c

More than 99 percent of natural thorium exists in the form of thorium-232 (Th-232). As a result of the radioactive decay of Th-232, other radioactive substances are produced. These include radium and radon. These substances give off radiation, including alpha and beta particles, and gamma radiation (ATSDR 1999).

A.3.2.1 *Thorium Mining*

Some rocks in underground mines contain thorium in a more concentrated form. After these rocks are mined, thorium is usually concentrated and changed into thorium dioxide or other chemical forms. After most of the thorium is removed, the rocks are called "depleted" ore or tailings. Thorium is used to make ceramics, gas lantern mantles, and metals used in the aerospace industry and in nuclear reactors (ATSDR 1999).

Domestic mine production of thorium-bearing monazite ceased in 1994 as world demand for ores containing naturally occurring radioactive thorium declined. Domestic demand for thorium ores, compounds, metals, and alloys has exhibited a long-term declining trend. No domestic thorium consumption was reported in the United States in 2005, according to the U.S. Geological Survey's canvass of mines and processors (USGS 2007c). Imports and existing stocks supplied essentially all thorium consumed in the United States in 2006.

A.3.2.2 *Thorium Use in a Nuclear Reactor*

Thorium can be used to produce fissile U-233 to fuel nuclear reactors. When Th-232 absorbs a neutron, it ultimately becomes U-233. The U-233 created in the reactor is a more effective fuel than U-235 or Pu-239 in a thermal neutron spectrum, and can provide a significant contribution to sustaining a reactor's operation.

Because thorium is a lighter element than uranium and plutonium, when thorium is used as a major component of reactor fuel, the production of transuranics (neptunium, plutonium, americium, and curium), the primary contributors to long-term waste toxicity and heat load in

geologic repositories, is reduced relative to conventional uranium-based fuels (IAEA 2002b). However, U-233 has a longer half-life than Pu-239 and can also contribute to long-term radiotoxicity (IAEA 2002b).

Enriched uranium is generally used as the start-up fuel[2] for a thorium-fueled reactor, where the U-235 would be enriched to approximately 19.9 percent (less than 20 percent, due to proliferation concerns) and a certain amount of U-238 would be inherently present. While this leads to some plutonium production (Todosow 2003), the quantity of the plutonium produced is significantly lower than in conventional uranium-based fuel, and distribution of the plutonium isotopes within the spent fuel is less attractive for potential use in a weapon. In addition, proliferation resistance is enhanced due to the presence of U-232[3] and its strong gamma emitting daughter products, and the fact that the thorium can be mixed with uranium initially to "denature"[4] the bred U-233 to keep its concentration below the accepted nonproliferation limits (IAEA 2005a).

The thermal conductivity of thorium is higher than that of uranium over a large temperature range. As a consequence, for the same power level, fuel-operating temperatures in a thorium-fueled reactor would be lower than those of a uranium-fueled reactor, and all thermally activated processes—such as diffusion of fission gas from the fuel—would be decreased. Alternatively, in a reactor fueled with thorium, reactor power could be increased, as could burnup, due to better fission product retention. Thorium dioxide (ThO_2) is chemically very stable and does not oxidize—a benefit for normal operation, postulated accidents, and in waste management (IAEA 2002b).

Several experimental and prototype power reactors were successfully operated during the mid-1960s through the 1980s using thorium fuels. In addition, the Indian Point-2 commercial pressurized water reactor (PWR) in New York successfully used thorium-based fuel, and thorium-based fuel was also used in several commercial HTGRs. However, despite the generally positive experience with these fuels, so far, thorium fuels have not been introduced commercially on a large scale, mainly because the estimated uranium resources have turned out to be sufficient to support the existing reactor fleets in a cost-effective manner.

The thorium open fuel cycle (see Figure A.3.2.2-1), while different in many aspects from the existing uranium once-through fuel cycle, can be characterized as a "new fuel design" rather than as a new reactor concept because it can be used with existing reactor technologies. In fact, based on recent studies, albeit generally not involving detailed designs, the thorium fuel cycle would be feasible for implementation in most existing commercial nuclear power plants without major modifications in the engineered systems (e.g., control rods and soluble boron control systems) (IAEA 2005a).

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[2] Plutonium could also be used as a start-up fuel.
[3] U-232 is formed when Th-232 absorbs a neutron, and then decays to U-232.
[4] In this context, the term "denature" means ensuring that there is enough other non-fissile isotopes of uranium (such as U-238) to maintain the percentage of U-233 low enough to mitigate concerns from a weapons-usability standpoint.

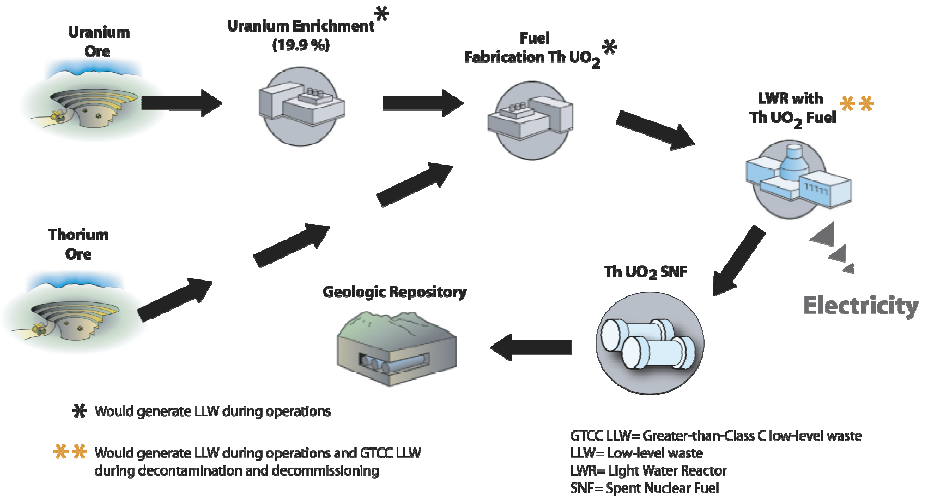


FIGURE A.3.2.2-1—*Thorium Open Fuel Cycle*

For purposes of this PEIS, only a thorium once-through fuel cycle as implemented in a LWR is assessed. While it is technically possible to recycle the SNF from a thorium-based fuel cycle, this is not assessed as a reasonable alternative in this PEIS for the reasons explained in Section 2.8.

A.3.3 Characterization of Commercial Nuclear Fuel Fabrication Facilities

Overview of United States Facilities

Table A.3.3-1 summarizes information about the fuel fabrication facilities in the United States. The table also includes information about a MOX fuel fabrication facility which has received U.S. Nuclear Regulatory Commission (NRC) authorization for design and construction and is currently undergoing the application process for an operating license (NRC 2007j). Construction of this MOX fuel fabrication facility began at DOE's Savannah River Site in 2007.

TABLE A.3.3-1—*Summary of Commercial Fuel Fabrication Facilities*

Licensee	Location
Uranium Fuel Fabrication Facilities	
Global Nuclear Fuel-Americas, LLC	Wilmington, NC
Westinghouse Electric Company, LLC	Columbia, SC
Nuclear Fuel Services, Inc.	Erwin, TN
BWX Technologies, Inc.	Lynchburg, VA
AREVA NP, Inc.	Lynchburg, VA
AREVA NP, Inc.	Richland, WA
Mixed Oxide Fuel Fabrication Facilities	
Shaw AREVA MOX Services (formerly Duke Cogema Stone and Webster)	Aiken, SC

Source: NRC 2007j

It should be noted that Nuclear Fuel Services, Inc., BWX Technologies, Inc., and AREVA NP, Inc. Lynchburg, Virginia, although NRC licensed, do not presently have the capability of processing UF_6 to UO_2 powder.

A.4 REACTORS

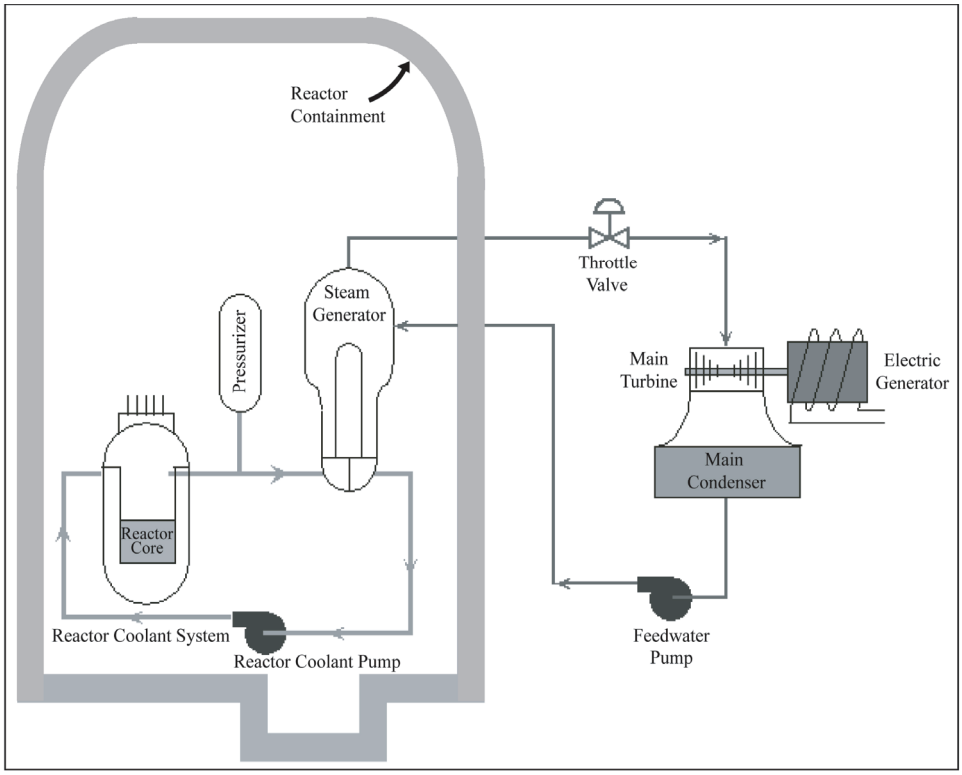
This section provides additional information regarding reactor technologies that could be deployed following decisions made as a result of this GNEP PEIS. As presented in Chapter 2 of the PEIS, the fuel cycle alternatives addressed in this PEIS could utilize various technologies, including different reactor types and reactor fuels.

Reactors can be categorized based on the neutron energy levels present in the reactor. A thermal spectrum reactor has low neutron energy and a fast spectrum reactor has high neutron energy, or simply "thermal" and "fast" for short. Most reactors in the world today are thermal reactors, and more specifically, light water reactors (LWRs) fueled with uranium-oxide (UO_2). Section A.4.1 discusses various LWR technologies and Section A.4.2 discusses the advanced recycling reactor, which is the representative fast reactor technology presented in the PEIS.

How does a typical U.S. commercial nuclear power plant work?

A nuclear power plant generates electricity using a "reactor," which is a device designed to use the fission process (splitting of atoms) to turn a small amount of mass into energy in a controlled way. In a typical nuclear power plant, neutrons strike uranium fuel, which splits, or fissions. Each fission produces energy, neutrons, and fission products; the energy produced takes the form of heat. The neutrons cause additional fission reactions. The heat from the fission reaction is removed from the reactor by a primary coolant which carries the heat to a steam generator. The steam generator uses the heat in the primary coolant to create steam in a secondary loop, which drives a steam turbine and produces electricity (see Figure A.4-1). Thus, in a nuclear power plant, the fission of nuclear fuel plays the same role as the burning of coal, natural gas, or oil plays in fossil fuel power plants.

Uranium is contained in fuel rods, which are clustered into fuel assemblies; each reactor core contains many fuel assemblies. During the operation of the reactor, the uranium in the fuel rods is being "used" and eventually enough fission products accumulate to interfere with the efficiency of the nuclear reaction such that the fuel can no longer effectively produce energy. At this point (typically 18 months), the used fuel is said to be "spent" and must be replaced with new (fresh) fuel assemblies in order for the reactor to continue to produce electricity. Approximately one-third of the reactor's fuel is replaced during each refueling.



Source: FPL 2008

FIGURE A.4-1—*Electricity Production in a Typical Commercial Nuclear Plant*

Several technologies are available for rejecting heat from operating large power generation facilities (such as nuclear reactors): once-through cooling; evaporative cooling; and dry cooling. The estimates for annual make-up water are bounded on the high-end by once-through cooling with water (Briggs et al. 2007). While not currently used for a nuclear power generation facility, based on successful use of dry cooling systems in large fossil power generating stations (Nagel and Wurtz 2006), a dry cooling system may be used in arid areas to reduce the amount of water used by the reactor facility. Dry cooling systems are fundamentally the same system used to cool an automobile engine. Process fluid (for example, steam) is on the inside of pipes or coils; heat is transferred from the fluid through the metal wall of the pipe or coil, into the surrounding atmosphere. In a dry cooling system, there is no direct contact between the fluid and the atmosphere. Because a dry cooling system does not use water's latent heat of vaporization for dissipating heat, the dry system would be larger than either a once-through or evaporative cooling system, and therefore would have a higher initial cost for an equivalent cooling capacity. Additionally, a reactor facility design would need to address environmental concerns such as noise associated with any design using fans, and esthetic factors like the large size of a dry cooling system compared with once-through or evaporative systems.

A.4.1 Thermal Reactors

LWRs fueled with UO_2 dominate world commercial nuclear power production because their technology is well known and has proven to be economical (see Section A.4.1.1). This section

also discusses mixed-oxide (MOX) fueled LWRs (Section A.4.1.2), HWRs (Section A.4.1.3), HTGRs (Section A.4.1.4), and thorium fueled LWRs (Section A.4.1.5).

A.4.1.1 *Uranium-Oxide Fueled Light Water Reactors*

LWR technology was initially developed in the United States. This technology now is the predominant nuclear energy technology used throughout the world. In fact, approximately 80 percent of the nuclear power plants operating in the world are LWRs (IAEA 2004a). About 34 power reactors are currently being constructed in 11 countries, most notably China, South Korea, Japan and Russia (WNA 2008b). Most of these reactors are LWRs.

All 104 nuclear plants used in the United States for electric power production are LWRs. The last new nuclear reactor to come on line in the United States was the Watts Bar Unit 1 reactor in Tennessee, owned and operated by the Tennessee Valley Authority. Its operating license was granted in February 1996 and it began commercial service in May 1996 (EIA 2008d).

LWRs are fueled with UO_2 fuel assemblies (Section A.3.1.4) that are enriched to approximately 3 to 5 percent U-235 (Section A.3.1.3). LWRs are classified as either PWRs or boiling-water reactors (BWRs), depending on whether the coolant water is kept under pressure or not. In a PWR, the pressurized water in the reactor creates steam by heating water in a steam generator (see Figure A.4-1). This steam turns a turbine to generate electricity. In a BWR, the reactor coolant boils and turns the turbine directly, without the need for a steam generator. In the United States, approximately 66 percent of the LWRs (69 out of 104) are PWRs and 34 percent (35 out of 104) are BWRs. Both PWRs and BWRs contain emergency cooling systems that would be utilized to maintain a safe reactor core temperature in the event of a loss-of-cooling situation.

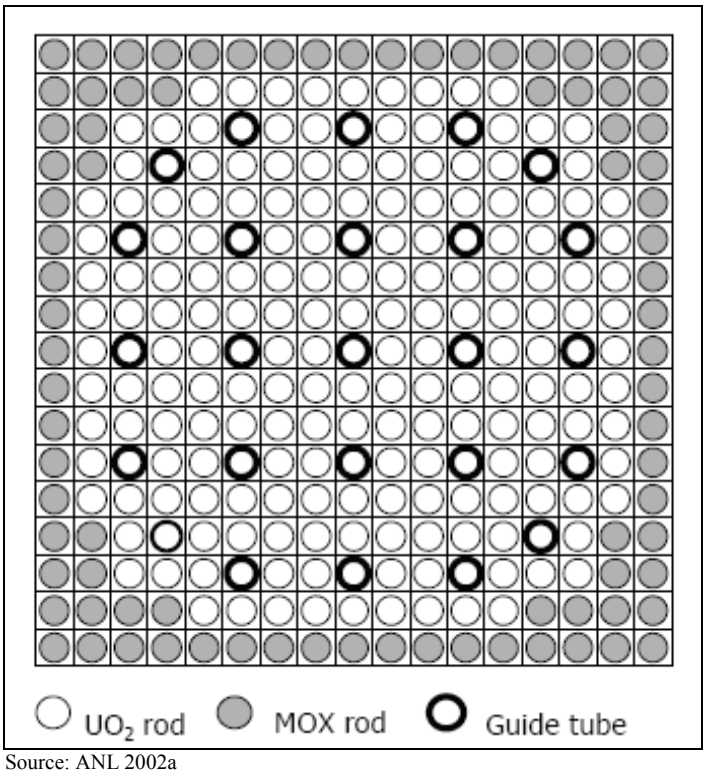
The commercial nuclear power industry is currently pursuing the development of advanced LWRs (ALWRs) which are designed to incorporate improved safety features such as passive systems. Passive systems (such as gravity and natural circulation) are intended to simplify safety systems and reduce costs, improve reliability, and mitigate the effect of human errors and equipment failures by increasing the time operators have to cope with accident conditions. Passive systems would also reduce reliance on offsite and onsite power supplies in the event of a loss of power. ALWRs are being developed over a large range of power levels, from small ALWRs (less than 300 megawatts electric [MWe]), medium ALWRs (300 to 700 MWe), and large ALWRs (greater than 700 MWe). The largest ALWR under development is approximately 1,600 MWe (IAEA 2004a). The No Action Alternative, Fast Recycle Alternative, Thermal/Fast Recycle Alternative, and the Thermal Recycle Alternative (option 2 and 3) would include LWRs/ALWRs fueled with uranium-oxide.

A.4.1.2 *Mixed Oxide Fueled Light Water Reactors*

The Thermal/Fast Recycle Alternative and the Thermal Recycle Alternative (Option 1) would involve thermal recycle in LWRs using a MOX fuel assembly concept. This fuel cycle would use a MOX fuel with either uranium-plutonium (MOX-U-Pu)[5] or uranium-transuranics (MOX-U-TRU) (Pu, neptunium [Np], americium [Am], and curium [Cm]). The MOX-U-Pu fuel is discussed in this section. The MOX-U-Pu fuel assembly, which could be fabricated using existing fuel fabrication technology, would partially replace the UO_2 fuel assemblies in LWRs. MOX fuel has been used extensively in the nuclear industry. MOX fuel was first used in a thermal reactor in 1963, but did not come into commercial use until the 1980s. So far about 2,200 tons (2,000 MT) of MOX fuel has been fabricated and loaded into power reactors. In 2006 about 198 tons (180 MT) of MOX fuel was loaded into over 30 reactors (mostly PWRs) in Europe. Today, MOX is widely used in Europe and is planned to be used in Japan. Currently about 40 reactors in Europe (Belgium, Switzerland, Germany, and France) are licensed to use MOX, and over 30 are doing so. Japan also plans to use MOX in up to 20 of its reactors. Most reactors use it as about one-third of their core, but some will accept up to 50 percent MOX assemblies. France aims to have all its 900 MWe series of reactors running with at least one-third MOX. Japan aims to have one-third of its reactors using MOX by 2010, and has approved construction of a new reactor with a complete fuel loading of MOX (WNA 2008c).

The assembly is made up of approximately two-thirds of UO_2 rods on the inside, and of approximately one-third of MOX-U-Pu rods on the periphery. When irradiated in a thermal reactor, the assembly would have an approximately zero plutonium mass balance (i.e., the plutonium consumed in the MOX-U-Pu rods would be approximately the same as the plutonium produced in the UO_2 rods). A MOX-U-Pu fuel assembly is displayed in Figure A.4.1.2-1.

[5] Neptunium could be added to the MOX-U-Pu fuel.



Source: ANL 2002a

FIGURE A.4.1.2-1—*Pin Loading Pattern of Mixed Oxide-Uranium-Plutonium Assembly*

The plutonium would be multi-recycled in the Thermal Recycle (Option 1) concept with the aim of stabilizing the plutonium inventory. Following discharge of the assembly, the MOX-U-Pu rods would be recycled to recover the U-Pu from the assembly. During the separation, most U-Pu (more than 99 percent) would be recycled, while all fission products and the minor actinides (Np, Am, Cm, and higher) would be separated during fuel reprocessing between recycle passes and sent to waste storage and eventual disposal in a repository. The reusable material would then be used for fabricating the fuel for the next stage of the multi-recycle operation. Figure A.4.1.2-2 shows the flow diagram for the Thermal Recycle Alternative (Option 1).

Each recycle stage would take about 11.5 years in the design currently being evaluated. The fuel would be resident in the core for 4.5 years. A 5-year cooling time is assumed, after which fuel separation and fabrication would take place. An additional 2 years is assumed before the fuel would be loaded back into the reactor core.[6] Each operational cycle would be 1.5 years and the fuel would accumulate approximately 15 gigawatt days (GWd) per metric ton (MT) of burnup in

[6] This additional 2 years is based on the realization that it takes time subsequent to fuel separation before the recovered fuel material is actually used in the reactor core. This time includes the elapsed time from fuel separation to fabrication into MOX fuel (which could occur at a different plant) and the re-introduction of the MOX fuel into a reactor core (at a different location). The times for transportation of materials/fuels between facilities and all associated wait times at the facilities are included. It is worth noting that each of the closed fuel cycle alternatives have more complicated logistics issues, compared to open fuel cycle alternatives, that could affect operational cycles.

this period. This would result in a discharge burnup of 45 GWd/MT for the MOX-U-Pu assembly (ANL 2007). In this concept, only the fission products, minor actinides, and Pu lost during reprocessing would pass to the repository.

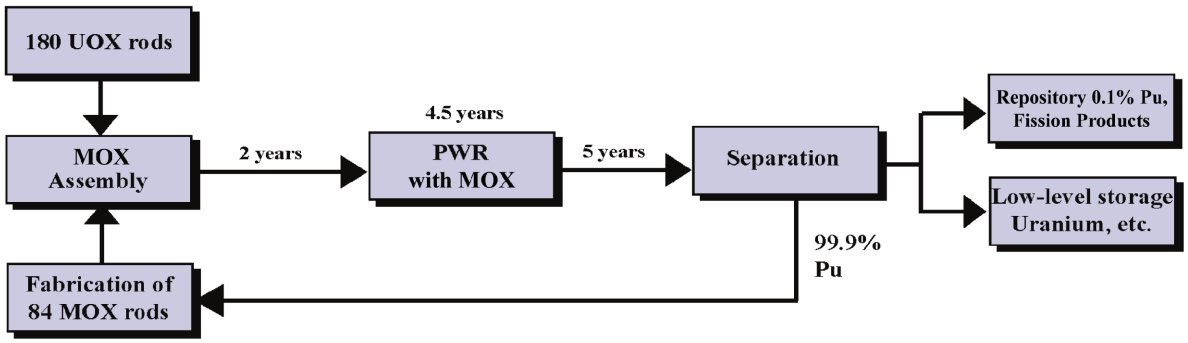


FIGURE A.4.1.2-2—*Multi-Recycling of Plutonium in Light Water Reactors*

A.4.1.3 *Heavy Water Reactors*

Two of the alternatives in the GNEP PEIS consider the use of HWRs: the Thermal Recycle Alternative (Option 2), which is also referred to as the "DUPIC (direct use of spent PWR fuel in CANDU) fuel cycle," and the HWR/HTGR Once-Through Fuel Cycle Alternative (Option 1—all HWRs). For each alternative, the HWR would operate the same. The major difference would involve the source of fuel for the HWR.

HWRs use deuterium-oxide (heavy water) as a moderator and coolant for their reactor core. Deuterium is a stable but rare isotope of hydrogen containing one proton and one neutron in its nucleus. Common hydrogen has only one proton in its nucleus. This makes deuterium twice as heavy as hydrogen. Heavy water has two deuterium atoms attached to an oxygen atom whereas natural water has two hydrogen atoms attached to an oxygen atom.

Chemically, the additional neutron in heavy water changes its characteristics only slightly, but in nuclear terms the difference is significant. The role of water as the moderator in a thermal reactor is to slow neutrons down to an energy level where they will cause fissions to occur in uranium atoms in the fuel. Since the natural water used in LWRs absorbs more neutrons than heavy water, LWR reactor fuel must be enriched to increase the amount of fissionable U-235 content needed to maintain a nuclear reaction. With fewer neutrons absorbed by heavy water (600 times fewer), more are available to fission with the uranium atoms in the fuel, and therefore, enrichment is not required. This enables natural uranium to be used for fuel in a HWR. For this reason, some proponents believe there is less risk of nuclear proliferation in a HWR. However, HWRs produce more weapons useable Pu-239 (from fertile U-238) because of the neutrons available, thereby offsetting the potential advantage of using natural uranium fuel (Miller 2001).

HWRs were first introduced in the United States in the early 1950s. Argonne National Laboratory (ANL) had two such reactors operating by 1950. These reactor designs served in large part as the design basis for the HWRs built at the Savannah River Site for the nuclear

weapons program. Five HWRs were brought on line at the Savannah River Site between 1953 and 1955. Heavy water allowed the United States to produce both plutonium and tritium for weapons using the same reactor technology. In 1993, the last of these reactors (K Reactor) was placed in cold standby, and later shut down (SRS 2007).

Canada has played a major role in the development of the HWR and has a number of heavy water power reactors in operation and under continued development. Canada began HWR development after World War II and built some early research reactors. In the 1950s, they began development of the CANDU power reactor concept. CANDU is a pressurized heavy water reactor using natural uranium fuel. The selection of this concept built upon the Canadians' previous experience and allowed them to utilize indigenous uranium reserves. The use of natural uranium avoids the requirement for uranium enrichment capability and eliminates the creation of depleted uranium enrichment plant tails, which create a waste management problem (Canada 2007, Whitlock 2000, Boczar et al. 2002).

Canada's 20 MWe Nuclear Power Demonstration (NPD) reactor went into operation in 1962 and successively larger plants were designed and built thereafter including the 500 MWe Pickering Station plants that went into operation in the early 1970s and the 800 MWe Bruce Station plants that went into operation in the late 1970s. Two models of the CANDU reactor have been marketed internationally; the CANDU-6 which has a capacity of approximately 700 MWe and the CANDU-9 with a capacity of approximately 900 MWe. The Advanced CANDU Reactor (CANDU-ACR) is a third generation reactor incorporating innovative features and improvements. The 750 MWe ACR-700 and the 1080 to 1200 MWe ACR-1000 are the main focus of current Canadian design efforts.

While natural uranium fuel is used in Canada, a variety of enrichments and fissile loadings can be accommodated in existing CANDU designs. These include slightly enriched uranium (SEU), mixed oxides of plutonium/uranium or plutonium/thorium, and fuels containing no fertile material.

The CANDU reactors are refueled at full power, a capability created by the subdivision of the core into hundreds of separate pressure tubes that contain fuel. Each pressure tube holds a single row of fuel bundles sitting end to end, 1.6 ft (0.5 m) long and weighing approximately 44 lbs (20 kg). The fuel bundles are surrounded by heavy water coolant within the pressure tubes, and the space between the pressure tubes is also filled with heavy water which serves as the reactor moderator. Several hundred pressure tubes form the fissile core within a cylindrical reactor vessel (the "calandria"). The calandria is horizontal, thus, making the CANDU core horizontal.

In order to refuel the CANDU, two refueling machines are used; one at each end of the core. One machine withdraws SNF as the second machine inserts fresh fuel from the opposite end. Six to 10 fuel bundles are shuffled each day of reactor operation. This reactor configuration and refueling scheme greatly reduces the cost of refueling. However, this reduction is offset by the need to periodically upgrade ("enrich") the heavy water moderator in the reactor. A purity decrease of only 0.1 percent can seriously affect the efficiency of fuel utilization in the reactor. Other advantages of on-power refueling include increased capacity factors, the ability to detect and remove defective fuel, and the ability to control power distribution in the core. The low

excess reactivity of the natural uranium fuel cycle allows the CANDU core to be continuously refueled rather than changed out in a batch process, like conventional LWRs.

India is developing a 300 MWe Advanced Heavy Water Reactor (AHWR) as the third stage in its plan to utilize indigenous thorium reserves to fuel its overall nuclear power program. The AHWR uses a combination of thorium/uranium-233 oxide fuel and plutonium/thorium oxide fuel to extend core life.

Historically, heavy water for most HWRs has been extracted from ordinary water, where deuterium occurs naturally at a concentration of about 150 parts per million (deuterium to hydrogen). As an isotope of hydrogen, the separation of deuterium from normal hydrogen is relatively simple, but deuterium's low natural abundance requires processing large volumes of water and makes such separation expensive. A vacuum distillation process is a simple, attractive process for producing small quantities of heavy water; however, multiple process steps and high energy consumption make this manufacturing approach unattractive for fulfilling the needs of multiple, large reactors (Miller 2001, Suppiah and Spagnolo 1998).

For many years, the primary extraction process used for bulk commercial production of heavy water in Canada was the Girdler-Sulfide chemical process. Using an "extraction tower," ordinary water was passed over perforated trays through which hydrogen sulfide gas was bubbled. By increasing temperature in one section of the tower, deuterium migrated to the hydrogen sulfide gas, and then in a cold section of the tower, migrated back into the water. Using multiple towers, water with increasing deuterium content was passed through successive stages and then through a vacuum distillation process until "enrichment" to reactor grade heavy water was achieved. The Girdler-Sulfide process supplied heavy water for the CANDU reactors built from the 1960s to the 1980s, although it was expensive and required large quantities of toxic hydrogen sulfide gas. This process was discontinued in 1997.

For the CANDU reactors, Canada has worked extensively over several decades on the development of industrial chemical processes that exhibit attractive economic characteristics in order to develop a more cost effective approach to separating deuterium. More advanced processes pursued by Canada have included the combined electrolysis and catalytic exchange process and the combined industrial reforming and catalytic exchange process. Although combined electrolysis and catalytic exchange can produce heavy water at roughly half the cost of vacuum distillation, by itself it is suitable for producing only small quantities of heavy water. The Canadians are now pursuing a three stage process to produce heavy water: the combined industrial reforming and catalytic exchange process followed by a bi-thermal water-hydrogen second stage, and finally, a combined electrolysis and catalytic exchange third stage to bring heavy water enrichment to reactor grade. These catalyst technologies are more environmentally benign than the gas extraction process they replace and are capable of producing heavy water at a reasonable cost.

In the case of the CANDU reactors, a fuel bundle consists of sintered uranium dioxide pellets in zirconium alloy tubes fabricated into a fuel bundle. Each fuel bundle is relatively small and lightweight with six components: ceramic pellets, zirconium alloy tubes (with a lubricant coating inside), spacer pads, bearing pads, end caps, and end plates. Each fuel bundle is roughly

20 inches (in) (50 centimeters [cm]) in length, 4 in (10 cm) in diameter, and weighs approximately 44 lbs (20 kg). The fuel used in CANDU reactors has typically been natural uranium fabricated into 37 identical fuel pins arranged about the long axis of the fuel bundle. The newer CANFLEX (CANDU FLEXible fueling) fuel has 43 fuel pins with two pin sizes. It is also about 4 in (10 cm) in diameter, 20 in (50 cm) long, and 44 lbs (20 kg) in weight. The two different pin sizes produce two different power outputs but the same overall bundle output. Special geometry modifications enhance heat transfer between the fuel and the surrounding coolant. The CANFLEX fuel has been designed to provide greater operating and safety margins, extended plant life, and better plant economics (Whitlock 2000, Boczar et al. 2002). The light weight of the CANDU fuel bundles (when compared to LWR fuel) and the use of natural uranium, simplifies manufacturing and handling of the fuel.

For each GWe-year of operation, a HWR produces approximately 53 metric tons heavy metal (MTHM) of SNF. While this mass is approximately 2.5 times greater than the SNF from a comparable LWR, the radiotoxicity of the HWR SNF is less (Canada 2007).

A.4.1.4 *High Temperature Gas-Cooled Reactors (HTGRs)*

HTGRs use graphite as a moderator to slow down neutrons and gas circulation to remove heat from the reactor core. The initial development of gas cooled reactors began at the Oak Ridge National Laboratory (ORNL) in 1943 with the air cooled 3.5 megawatts thermal (MWth) X-10 reactor. Power reactor development for purposes of producing electricity began in Great Britain in the 1950s where carbon dioxide was used as a coolant. Magnox reactors were the first plant concepts built and operated in Britain. Twenty-six reactors of this design were constructed and the first of these, Calder Hall, was the first commercial nuclear power plant in the world. Two additional reactors were sold outside of Britain; one to Italy and one to Japan. Four of the British units remain in operation today, but are expected to be shutdown by 2010. Great Britain also developed the advanced gas-cooled reactor (AGR) after the Magnox reactors were deployed. In 2007, there were approximately 14 AGRs in operation (IAEA 2007a, IAEA 2007b).

The development of helium cooled gas reactors began in the 1960s with prototype power plants constructed in the United States, Great Britain, and Germany. Helium coolant allows the gas reactor to achieve higher operating temperatures and therefore higher efficiencies. The 13 MWe AVR (Arbeitsgemeinschaft Versuchsreaktor) in Germany operated successfully for 21 years demonstrating the application of HTGR technology for electric power production. The 300 MWe Thorium High Temperature Reactor (THTR-300) was another plant built and operated in Germany which helped demonstrate the HTGR concept. Both were pebble bed reactors that used U-235 and Th-232 fuel. Pebble bed reactors are fueled by spheres of graphite moderator with small particles of fuel dispersed throughout. These spheres are stacked in a close packed lattice and cooled by helium. The heated helium may then be used to create steam for electricity or drive a turbine generator directly.

HTGR reactors built and operated in the United States included the 40 MWe Peach Bottom plant in Pennsylvania, which went online in October 1974, and the 330 MWe Fort St. Vrain plant in Colorado, which operated from December 1973 to August 1989. Peach Bottom used a round graphite tube arrangement containing fuel particle compacts made from thorium and uranium

fuel. Fort St. Vrain had a hexagonal (prismatic), graphite block core with thorium and uranium fuel. The fuel in a prismatic core is made of small particles pressed into graphite compacts that are placed into the graphite blocks (Ft. St. Vrain 1). All of the German and U.S. HTGR plants are now shut down.

More recent plants still in operation include the 30 MWth High Temperature Engineering Test Reactor (HTTR) in Japan which reached full power in 1999, and the 10 MWth High Temperature Gas Cooled Reactor (HTR-10) in China that achieved first criticality in 2000. The HTTR uses a hexagonal graphite block core like the U.S. designs. Japan is considering construction of a larger (300 MWe) reactor design (the Gas Turbine Helium Reactor [GTHR]-300) for hydrogen production. China is also considering construction of a larger reactor in the 300 to 400 MWth range for electricity production, district heat production, and generation of process heat. The Chinese reactor would be a pebble bed design like their HTR-10 reactor and use either a steam or direct cycle turbine for electricity production.

A pebble bed modular reactor (PBMR) is being developed for commercial use by an international conglomerate that includes South African based ESKOM (a public utility established by the South African government in 1923). The initial objective of the conglomerate was to design and build a 116 MWe PBMR demonstration reactor in Kroeberg, South Africa around mid-2005. The design has evolved to the point where the preferred module size is now approximately 400 MWth, considerably larger than the original concept size. The PBMR is based on the German THTR-300 design, but updated to include passive safety features and modified to drive a Brayton cycle gas turbine. Construction of a single unit prototype demonstration reactor at Kroeberg is now scheduled to begin in 2009 with fuel loading in 2013. The first commercial units could start construction as early as 2016 (Gee 2002, Hargraves 2007).

The Netherlands has conducted studies on a small, simplified version of the South African PBMR called ACACIA (Advanced Atomic Cogenerator for Industrial Applications) for the purpose of heat and power cogeneration. Their reactor is a 40 MWth pebble bed concept with a direct cycle helium turbine that is designed to produce 13.6 MWe. Several institutes and design agencies in the Netherlands, Germany, France, and other European countries are working together in a European HTGR program which is currently focused on fuel development, materials development, and licensing aspects of the HTGR.

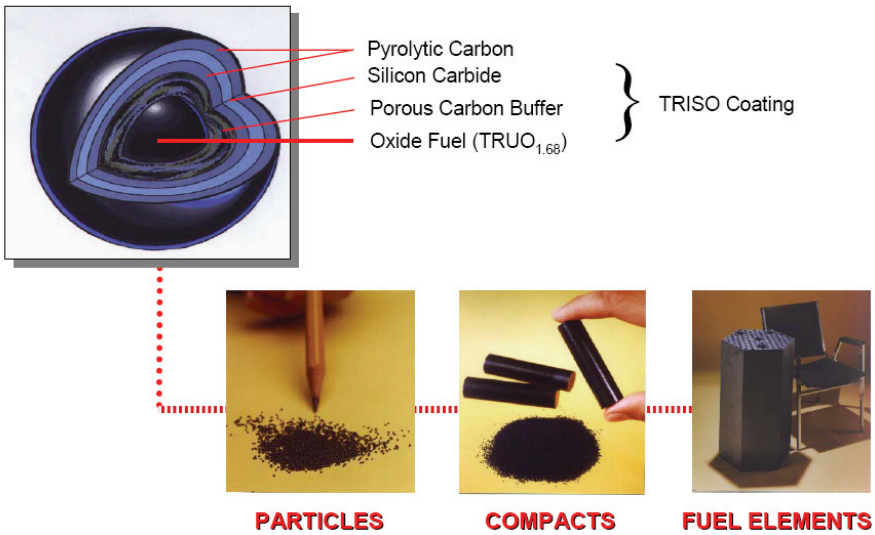
The Republic of Korea has shown interest in high temperature applications of HTGR technology; specifically hydrogen production. They are pursuing work on HTGR fuel technology.

DOE has focused substantial resources over the past decade on the Generation IV Nuclear Energy Systems Initiative (Generation IV) wherein new reactor systems are being developed for deployment over the next 20 years. The Next Generation Nuclear Plant (NGNP) is planned to be an advanced nuclear reactor design that can improve upon the current generation of operating commercial nuclear power plants. In addition to producing electricity safely and economically, the NGNP will focus on establishing the feasibility of producing both electricity and hydrogen from a nuclear reactor. DOE is considering the Very High Temperature Reactor (VHTR), an HTGR, as a candidate reactor technology under the NGNP program.

HTGRs, under the Thermal Recycle (Option 3), are also being considered for the deep-burn (relatively high consumption) of non-uranium based, transuranic fuel derived from LWR SNF, and a representative system is the deep burn modular helium reactor (DB-MHR) concept being developed by General Atomics (Kim et al. 2006, Hong et al. 2007). Under this concept, LWR spent fuel would be recycled in a nuclear fuel recycling facility to remove 99.9 percent of the actinides and 97 percent of the fission products. The removed actinides would then be used to manufacture fuel for a DB-MHR. For this concept, approximately 85 tons (77 MT) of spent LWR fuel would be required to manufacture 1.1 ton (1 MT) of fuel for the DB-MHR (ANL 2005). The essential feature of the concept would be the use of coated fuel particles (known as "TRISO") that are considered strong and highly resistant to irradiation (see Figure A.4.1.4-1). This would enable the fuel to remain in the reactor core for a long irradiation time (a burnup rate of more than 500 GWd/MTHM) which would cause a relatively high destruction of the transuranic isotopes by fissions from thermal neutrons. Recent evaluations have indicated that a transuranic destruction level as high as approximately 60 percent is attainable in a single-pass in the DB-MHR system (Kim et al. 2006).

The recycle of spent HTGR fuel in fast reactors is an alternative considered but eliminated from detailed study in this PEIS. Possible approaches include using an additional recycle pass in the DB-MHR (found ineffective) or passing the spent fuel transuranics of the DB-MHR to a fast spectrum system for additional irradiation in a closed fuel cycle. Both cases would require the development of a technology to recycle the TRISO fuel.

Detailed core and plant design and safety studies are needed to confirm the feasibility of the DB-MHR. (One safety attraction is that the DB-MHR response is stable to coolant loss, contrary to LWRs with high transmutation fuel loading.) Additional research and development (R&D) is also required to develop the non-uranium, transmutation fuel to ensure acceptable performance during irradiation. Other practical issues include the durability of the DB-MHR SNF in a repository setting, and the potentially large quantity of SNF material if fuel blocks are to be disposed in a geologic repository.



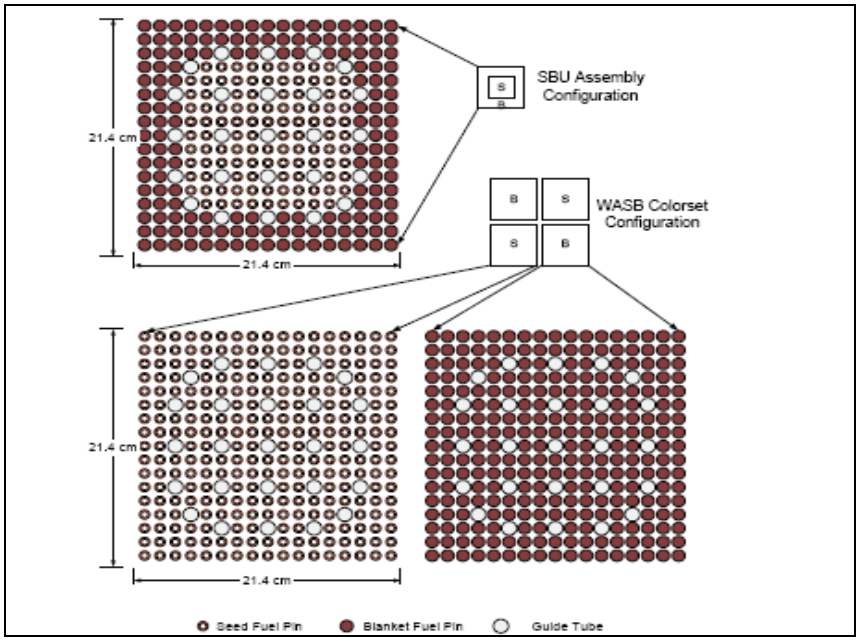
Source: ANL 2005

FIGURE A.4.1.4-1—*TRISO Fuel Concept Showing the Fuel Particles, Fuel Compacts, and Fuel Assembly Blocks*

A.4.1.5 *Thorium-Fueled Light Water Reactors*

There are a variety of thorium-based fuel concepts that have been previously studied for use in existing LWRs.[7] The basic idea would be to replace the standard/conventional UO_2 fuel assemblies in the LWR with new fuel assemblies that would utilize enriched uranium and thorium. Because thorium does not have any fissile isotopes, its introduction into a reactor must be compensated for by uranium enriched to greater than the conventional level for LWRs (less than 5 percent), but still below the threshold for highly enriched uranium (20 percent). Also, the fact that uranium-233 (U-233) would be bred from the thorium requires that additional uranium be present to "denature"[8] the bred U-233 to keep its concentration below the accepted non-proliferation limits (IAEA 2005a).

Two options that have been examined under the Nuclear Energy Research Initiative funded by DOE appear to offer potential benefits. The two approaches are: the seed-blanket unit, which employs a seed-blanket unit that is a one-for-one replacement for a conventional LWR fuel assembly; and the whole assembly seed and blanket, where the seed and blanket units each occupy one full-size LWR assembly and the assemblies are arranged in the core in a modified checkerboard array (See Figure A.4.1.5-1) (IAEA 2005a).



Source: IAEA 2005a

FIGURE A.4.1.4-5—*Seed-Blanket Unit and Whole Assembly Seed and Blanket Fuel Assembly Design*

[7] See, for example, Todosow 2003: "Use of Thorium in Light Water Reactors," M. Todosow, Brookhaven National Laboratory, *Advances in Nuclear Fuel Management III (ANFM 2003),* Hilton Head Island, SC, USA, October 5-8, 2003, on CD-ROM, American Nuclear Society, La Grange Park, IL (2003).

[8] In this context, the term "denature" means ensuring that there are enough other non-fissionable isotopes of uranium (such as U-238) to maintain the percentage of U-233 low enough, compared to the total uranium, so that the U-233 would not be a concern from a weapons-usability standpoint.

Full-core implementation of the seed-blanket unit and whole assembly seed and blanket concepts has been considered. All of the fuel assemblies in the reactor are replaced with seed-blanket units or, in the whole assembly seed and blanket approach, with seed or blanket assemblies. In both concepts, the seed and blanket represent approximately 40 and 60 percent of the fuel area, respectively. Because there are distinct thorium-bearing (blanket) and uranium-bearing (seed) regions, separate fuel management schemes for these zones are possible to optimize performance. This approach allows for the efficient generation and in-situ consumption of U-233 in the blanket region of the reactor, with the seed supplying neutrons to the blanket in the most efficient way (i.e., with a minimal investment of uranium). Because the generation of U-233 in the blanket is relatively slow, a long residence time for the blanket is needed for optimal performance. Therefore, the blankets are typically assumed to remain in the core for 6 to 9 fuel cycles. With these long residence times, the blanket will achieve high burnup[9] (on the order of 100 to 150 GWd/MTHM). While the properties of thorium oxide and initial fuel performance analyses suggest that this is possible, acceptable fuel performance has yet to be demonstrated. The seed material would need shuffling similar to conventionally fueled enriched uranium reactors. For each cycle, a third of all seeds in the core would be replaced by fresh seeds, while the remaining two-thirds of the partially depleted seeds would be shuffled to maintain an acceptable power density distribution (IAEA 2002b).

In a seed-and-blanket core, the seeds must sustain power levels that are significantly above average, while the blanket assemblies experience less stressful conditions. Thus, the design of the fuel in the seed rods, and the cooling, must take this into account. These demands can be accommodated in various ways—for example, by allowing more coolant to flow through the seeds, by making the fuel materials less resistant to the flow of heat (increasing the thermal conductivity), or by modifying the fuel geometry. Seed-blanket concepts offer the possibility of producing up to approximately 40 percent of the power by the fission of U-233. This open fuel cycle concept for introducing thorium in nuclear power reactors is very attractive from the point of view of 'in situ' utilization of U-233 and avoiding the handling of U-233 outside of the core (IAEA 2005a)[10]. Initial studies have shown that safety and operational parameters and performance would be comparable to those of existing power plants; however, additional studies are needed to demonstrate this (IAEA 2002b). The fuel design would be based primarily on an existing (not necessarily commercial) fuel technology. The maximum allowable initial fuel enrichment would be kept below the non-proliferation limit of 20 percent of the U-235 content, and the combined U-233 and U-235 content in the blanket would also be kept below this non-proliferation limit.

For the higher burnups of the seed-blanket unit approach, analyses of SNF storage and disposal requirements indicate that the discharged fuel mass could be reduced by approximately

[9] Burnup is a measurement of the fissile material consumed via fissioning during fuel irradiation. It is normally quoted in either megawatt days per kilogram (MWd/kg) or gigawatt days per metric ton heavy metal (GWd/MTHM). Typical fuel assemblies in an LWR remain in the core for 3 to 4 cycles and achieve a burnup of approximately 45 to 51 GWd/MTHM.

[10] A special feature of the thorium fuel cycle is the high gamma dose associated with the daughter products of U-232, which is always associated with U-233 and the high specific radioactivity of U-233. Hence, handling of thorium-based SNF require remote and automated operation in hot cells or shielded gloveboxes.

50 percent (IAEA 2002b, Todosow 2003). This reduction indicates a potential for a reduction in the overall SNF storage requirements and associated costs. The total amount of plutonium that would be produced, annually, would be reduced by approximately 3 to 5 times, relative to the existing uranium-based fuel cycle (IAEA 2002b), and the plutonium isotopes would be less attractive as material for potential use in a weapon (IAEA 2005a). The use of Th-232 and highly enriched uranium (HEU) also produces fewer minor actinides than conventional low-enriched uranium (LEU) fuel, thereby reducing the radiotoxicity for the first several thousand years (IAEA 2002b).

A.4.2 Fast Reactors

In a fast reactor, the neutrons produced by fission are not slowed down significantly before they cause more fission reactions. This contrasts with thermal reactors (e.g., light water reactors, heavy water reactors, and graphite moderated reactors) in which water or graphite is used to slow (or "thermalize") neutrons. Thermal neutrons are more-efficient at producing a fission reaction in a limited number of "fissile" isotopes, whereas the higher-energy (fast) neutrons can fission all types of uranium and transuranic elements. This allows a fast reactor to consume or "transmute" the transuranics while generating electricity. This section describes the sodium-cooled fast reactor that is the representative fast reactor technology analyzed in this GNEP PEIS, elsewhere, this PEIS refers to this technology as the advanced recycling reactor.

A.4.2.1 *Sodium-Cooled Fast Reactor*

The sodium-cooled fast reactor is the "fast reactor" technology that has been developed and demonstrated by operational experience in the United States and other countries around the world for over 50 years. Over 25 of these reactors have been built and successfully operated for research purposes and many have produced electricity. Several reactors of this type outside the United States are still in operation. Safe, stable and predictable operation of liquid metal cooled fast reactors has been demonstrated worldwide, resulting in an understanding of the necessary safety requirements, design features, and operating practices.

Although some of these reactors encountered problems including component and fuel failures and coolant fires[11], none of these problems had an impact to the public. In no case has there been an uncontrolled release of radioactive material, nor has there been any incident that resulted from a fundamental flaw in the concept of the liquid metal cooled fast reactor. Most of the reactors operated successfully and all provided a technology and experience base upon which new fast

[11]Two notable sodium cooled fast reactor accidents occurred. The first event took place on October 5, 1966, when the Fermi 1 reactor in Monroe County, Michigan, suffered a partial nuclear core meltdown. The accident was attributed to the interruption of cooling to a portion of the reactor core by a metal plate that had broken loose from a different part of the reactor. Two of the 105 fuel assemblies were partially melted and an additional four assemblies were damaged (GSU 2007). The second event occurred at the MONJU fast breeder reactor in Japan on Dec. 8, 1995, when a temperature measuring device in a pipe carrying sodium broke due to vibration. A subsequent sodium leak and fire occurred. The failure to detect the fire early led to significant sodium leakage and damage to the room where the failure occurred. The accident was caused by a design fault and made worse by the absence of an effective alarm system to notify plant operators (WISE Paris 2007). In the case of the Fermi 1 accident, no radiation was released off site. In the case of the MONJU accident, the leak was not in a radioactive system and therefore no radiation was released. Other reactor accidents, most notably the Chernobyl reactor accident in Ukraine and the Three Mile Island Accident near Harrisburg, Pennsylvania, were water cooled reactor accidents and provide extremely limited applicability to liquid metal reactor designs.

reactors can be designed, built and operated safely. Advanced fast reactor designs continue to be developed by the United States and foreign countries.

Seven liquid metal fast reactors were built and operated in the United States beginning in the early 1950s. An eighth reactor project, the Clinch River Breeder Reactor Project (CRBRP), was initiated in the early 1970s but then terminated in 1983 due to a lack of economic need at that time and the proliferation concerns that were associated with the breeder reactor. Subsequent projects by DOE, including the Integral Fast Reactor project which focused on development of an advanced metallic fuel for the sodium-cooled fast reactor and the Advanced Liquid Metal Reactor which supported the development of the Power Reactor Innovative Small Module (PRISM) liquid-metal reactor by General Electric, resulted in advances in understanding of the characteristics of sodium-cooled fast reactors. By the end of these projects, it appeared feasible to design a sodium-cooled fast reactor that would meet all regulatory and environmental requirements for an advanced nuclear reactor, as evidenced by the PRISM Mod B design that was reviewed by the NRC. Remaining issues were mainly on design and operational choices, not on the fundamental concept of a sodium-cooled fast reactor, as discussed in the Pre-application Safety Evaluation Report (NRC 1994a).

A.4.2.2 *Plant Description*

A fast reactor is conceptually similar to a thermal reactor. Both fast and thermal reactors have a reactor core where fuel is used to generate heat by fission. Coolants are used to transfer heat from the reactor core to turbines for electricity generation. A significant difference for the fast reactor is the use of liquid sodium as the coolant for the core. As mentioned previously, coolants like water tend to slow the neutrons. As a result, fast reactors must use a coolant that does not thermalize neutrons and which has sufficient heat transfer capability. A number of high-heat transfer coolants, such as liquid sodium/potassium metal, liquid lead, and liquid lead/bismuth, may be used for this purpose. Even gas has been proposed as a fast reactor coolant. Several of these coolants continue to be studied worldwide however; liquid sodium still appears to be the best choice for a fast reactor coolant.

A significant difference for the sodium-cooled fast reactor is the use of an ‘intermediate coolant loop’, as shown in Figure A.4.2.2-1. One purpose of the intermediate loop is to isolate the primary coolant loop inside of the reactor vessel from the water and steam loop, because water and sodium react energetically if they come into contact. Under normal circumstances, the piping in the steam generator separates the sodium and water. The use of an intermediate loop is to provide a safety barrier in the event of a failure of the piping in the steam generator so that consequences of a leak in the steam generator piping would be confined to the steam system and the intermediate loop and would not affect the primary loop cooling the reactor core. The intermediate loop also reduces the impact of the activation of the primary sodium coolant, so that activated (radioactive) sodium coolant is retained within the primary vessel and is not transported to other parts of the plant.

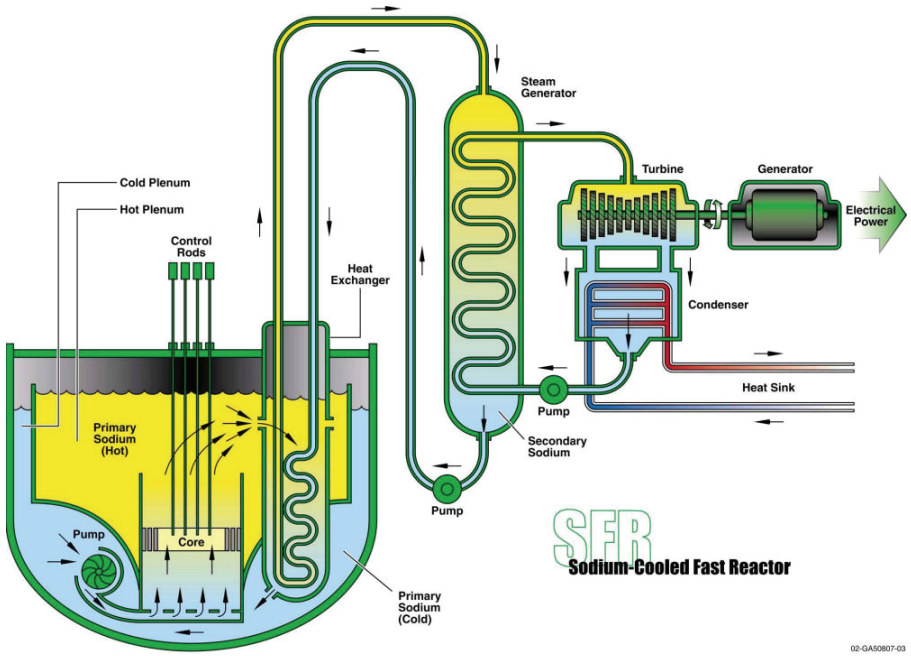


FIGURE A.4.2.2-1—*Schematic of a Sodium-Cooled Fast Reactor (Pool-type)*

Figure A.4.2.2-1 shows an example of what is called a 'pool-type' reactor because the reactor core and the entire primary coolant loop are contained in a large vessel using a pool of liquid sodium. This design is possible for a fast reactor because, unlike other reactors, the primary coolant loop of a sodium-cooled fast reactor operates at or near atmospheric pressure. A fast reactor can also be designed to be a 'loop-type' reactor, in which the primary loop reactor components are all connected by piping, as is done for light-water reactors, or as a combination of the two types. At this time it appears that it would be possible to design a successful sodium-cooled fast reactor with any of these design approaches.

A.4.2.3 *Reactor Fuel*

Fast reactor fuel is not necessarily the same as that used in a light water reactor. Although fast reactor fuel may be in oxide-form, it alternatively may be in metal, carbide, and nitride-form. If uranium fuel were used in a fast reactor, higher enrichment than for fuel used in a thermal reactor would be required, which would result in significant additional costs for preparing the fuel. Instead, fast reactors have been typically designed to operate with a fuel that is a mixture of natural uranium with about 15–30 percent plutonium. In the past, the focus had been on using the fast reactor to 'breed' more plutonium from natural uranium. In a "breeder reactor," more nuclear fuel is produced (by "transmuting" natural uranium) than was consumed in running the reactor. As a result, there is more plutonium in the reactor than was used in the fuel initially, essentially making more usable fuel. More recently, the effort has been on a different use for the fast reactor: to "consume" the transuranic materials from spent nuclear fuel to generate electricity as part of a closed fuel cycle. Fuel for this type of fast reactor is composed of a mixture of natural uranium and approximately 20–35 percent transuranic elements. The transuranic content may be higher depending on the desired rate of consumption of the transuranic elements. This configuration of a fast reactor is commonly referred to as a 'burner'

reactor because it does not produce more fuel than it uses, and allows the fast reactor to "consume" transuranic elements, which are typically a major concern in the disposal of spent nuclear fuel.

Fast reactor fuel is contained in fuel pins with metallic cladding in a manner similar to light water reactor fuel, but the cladding is usually stainless steel instead of zircaloy. Groups of fuel pins are housed in stainless steel ducts of hexagonal cross-section called "hexcans." A hexcan with fuel pins inside is the fuel assembly for the fast reactor. The hexcans are used to channel coolant flow to all of the fuel pins in the assembly. The reactor core is constructed from a large number of these fuel assemblies. The number of fuel assemblies required for a fast reactor is determined by the desired reactor power output.

The amount of SNF produced from reactor operations is a function of cycle length, or time between shutdowns for refueling. In large fast reactors (typically greater than 1,000 MWth), the cycle length would be targeted at 1 year in order to minimize plant down time and increase the amount of time the reactor is operating (capacity factor). The target cycle length would be optimized based on the function of the reactor. An advanced recycling reactor, for example, could be designed primarily to optimize destruction of transuranic elements. This would be different from current LWRs, which are optimized for power production.

A.5 REPROCESSING TECHNOLOGIES

This section presents technical descriptions of various SNF separation processes that could be candidates for future use to separate LWR and fast reactor SNF. The two primary SNF separation processes, aqueous and electrochemical, are described.[12] For further comparison information on aqueous and electrochemical processing, please refer to the WSRC 2008 Report (WSRC 2008a).

A.5.1 Aqueous Processing

Aqueous chemical treatment of various types of SNF has been practiced on a commercial scale in a number of countries. For example, PUREX was developed by the United States in the late 1950s and is in active use on a commercial scale in France, Japan, Russia, and the United Kingdom. However, as previously mentioned, the PUREX process separates pure plutonium, which is a proliferation concern. One of the objectives of AFCI is to develop, demonstrate, and deploy advanced technologies for recycling SNF that do not separate plutonium, with the goal over time of ceasing separation of plutonium and eventually eliminating excess stocks of civilian plutonium and drawing down existing stocks of civilian SNF (DOE 2007l). Examination of different aqueous processing technologies will include the requirement that SNF should only be

[12]In addition to the PUREX, COEX, NUEX, OREOX, and UREX processes, other processing technologies exist including: 1) Supercritical CO_2 which is a solvent extraction method that employs supercritical CO_2 with tributyl phosphate (TBP), similar to PUREX but without the organic diluent; and 2) UNEX which is the Universal Extraction process designed to remove all of the most troublesome radioisotopes (Sr, Cs, and minor actinides) from the HLW left after the extraction of uranium and plutonium.

processed using a technology, consistent with U.S. policy, that would not result in a civilian nuclear fuel cycle producing separated plutonium.

Aqueous processing generates several types of waste streams. Gaseous wastes are generated from shearing and dissolution, vessel vents, and high activity liquid waste vitrification. Gaseous emissions must be treated and resulting wastes disposed of properly. Treatments include, but are not limited to, NO_x scrubbing columns, iodine filters, mist filters, HEPA filters, condensers, and ruthenium absorption columns. The type of treatment depends on the gas composition.

Liquid LLW would also require treatment. Liquid LLW consists of low activity evaporator overheads, washroom water, laundry drains, equipment drains, water inside casks, floor drains, etc. The treatment processes used could include evaporation, solvent removal, filtration, solidification, and desalination. Once in a solid form, this LLW would be packaged in accordance with all applicable requirements and transported to an appropriate LLW management or disposal facility.

Liquid HLW must be treated to a solidified, leach-resistant waste form suitable for disposal in a geologic repository. Storage and cooling of the solidified HLW package, and packaging in accordance with all applicable requirements would be necessary prior to shipment to a geologic repository. To address concerns prompted by historical releases of liquid HLW from underground storage tanks, DOE would not support any long-term storage of liquid HLW.

In addition to the waste streams mentioned above, additional solid wastes would be generated as a result of aqueous processing. Greater-Than-Class-C (GTCC) low-level wastes, including those containing transuranics, may require treatment prior to packaging for disposal in a facility designated for such wastes. Other waste streams that would require management are solid LLWs, hazardous wastes, sanitary waste streams, and industrial waste streams.

PUREX

All of the currently operating reprocessing facilities use the PUREX process. The PUREX process (Figure A.5.1-1) is a proven technology that has been used by DOE since the 1950s and that has been used by U.S. commercial industry;[13] however, it does not meet the GNEP program proliferation resistance goal because it separates pure plutonium.

[13] See Chapter 1 for a discussion of the history of reprocessing in the United States.

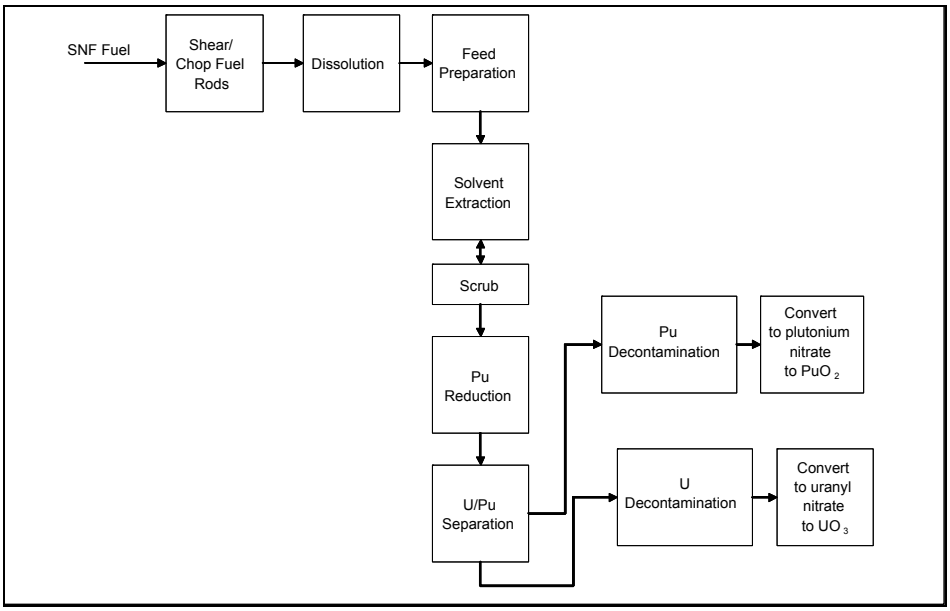


FIGURE A.5.1-1—*Plutonium Uranium Reduction and Extraction Flow Diagram*

The PUREX process was modified for the Japanese facility (Rokkasho), which is undergoing operational tests. In this process, some of the uranium product is blended with the plutonium (at a nominal 50:50 ratio) before the MOX product is sent to the fuel fabrication facility. The remaining uranyl nitrate solution produced by the PUREX process is converted to UO_3 for storage (Figure A.5.1-2).

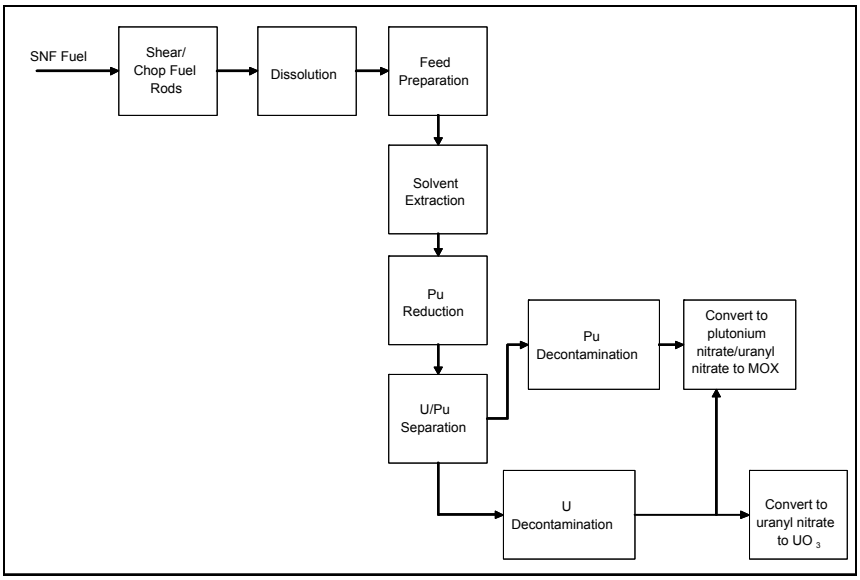


FIGURE A.5.1-2—*Rokkasho Flow Diagram*

COEX® and NUEX®

The French have developed a uranium-plutonium co-extraction process, COEX®, which is similar to the Rokkasho process, but does not produce a separated plutonium stream anywhere in the process line.

NUEX® is a proprietary co-extraction technology developed by the British, and licensed to EnergySolutions, Inc. Like COEX®, NUEX® produces a uranium-plutonium product stream and has no separated pure plutonium anywhere in the process line. It also separates the minor actinides.

Because both the COEX® and NUEX® processes are proprietary technologies, less information is publicly available for these separation processes than the PUREX and Uranium Reduction and Extraction (UREX) technologies presented in this section. However, because the COEX® and NUEX® processes use similar processes to PUREX and UREX, the environmental impacts (e.g., emissions, radiation dose to workers, and wastes) would not be expected to differ significantly.

UREX

In the past few years, DOE has made significant advancements in SNF processing technology R&D that has both important environmental and proliferation-resistant advantages over PUREX. This processing technology is known as UREX+ (for uranium extraction).

In the UREX+ process, plutonium, other transuranics, and fission products are extracted together in a single stream from which the transuranics could be extracted for reuse in nuclear fuel. The UREX+ reprocessing flowsheet differs from the PUREX reprocessing flowsheet in that plutonium is never isolated in the UREX+ scheme as it is in the PUREX scheme. Other actinide elements are kept with the plutonium and do not enter the HLW waste stream. This makes the material more difficult to use in a weapon. Additionally, because UREX+ does not place these actinides in the waste stream, there could be a reduction in the amount and activity levels of the HLW produced.

The analysis in this PEIS is based on a variation of the UREX+ process referred to as UREX+1a. Although UREX+1a has not been used on a large scale, it is used as a representative process in this PEIS because it draws upon a wider body of knowledge available on aqueous processing, and incorporates advanced separations technology features based on UREX+1a R&D work done to date. It also does not result in the separation of pure plutonium. The UREX+1a process leads to: a uranium stream, transuranic streams comprised of plutonium and other transuranic elements, and other streams comprised of fission products. Table A.5.1-1 lists other UREX+ separation variations. Depending on the number and types of separation processes employed, different products can result. For example, UREX+2 adds a step to separate plutonium and neptunium from the other transuranics (americium and curium). There is also an option to co-extract uranium and plutonium in one step instead of several extraction steps.

TABLE A.5.1-1—*Uranium Reduction and Extraction Separations Processes*

Process	Product 1	Product 2	Product 3	Product 4	Product 5	Product 6	Product 7
UREX+1	U	Tc	Cs/Sr	TRU+Ln	F.P.		
UREX+1a	U	Tc	Cs/Sr	TRU	All F.P.		
UREX+2	U	Tc	Cs/Sr	Pu+Np	Am+Cm+Ln	F.P.	
UREX+3	U	Tc	Cs/Sr	Pu+Np	Am+Cm	All F.P.	
UREX+4	U	Tc	Cs/Sr	Pu+Np	Am	Cm	All F.P.

Source: WSRC 2007d
Note: U = uranium; Tc = technetium; Cs/Sr = cesium/strontium; TRU = transuranics; Pu = plutonium; Np = neptunium; F.P. = fission products; Am = americium; Cm = curium; Ln = lanthanides.

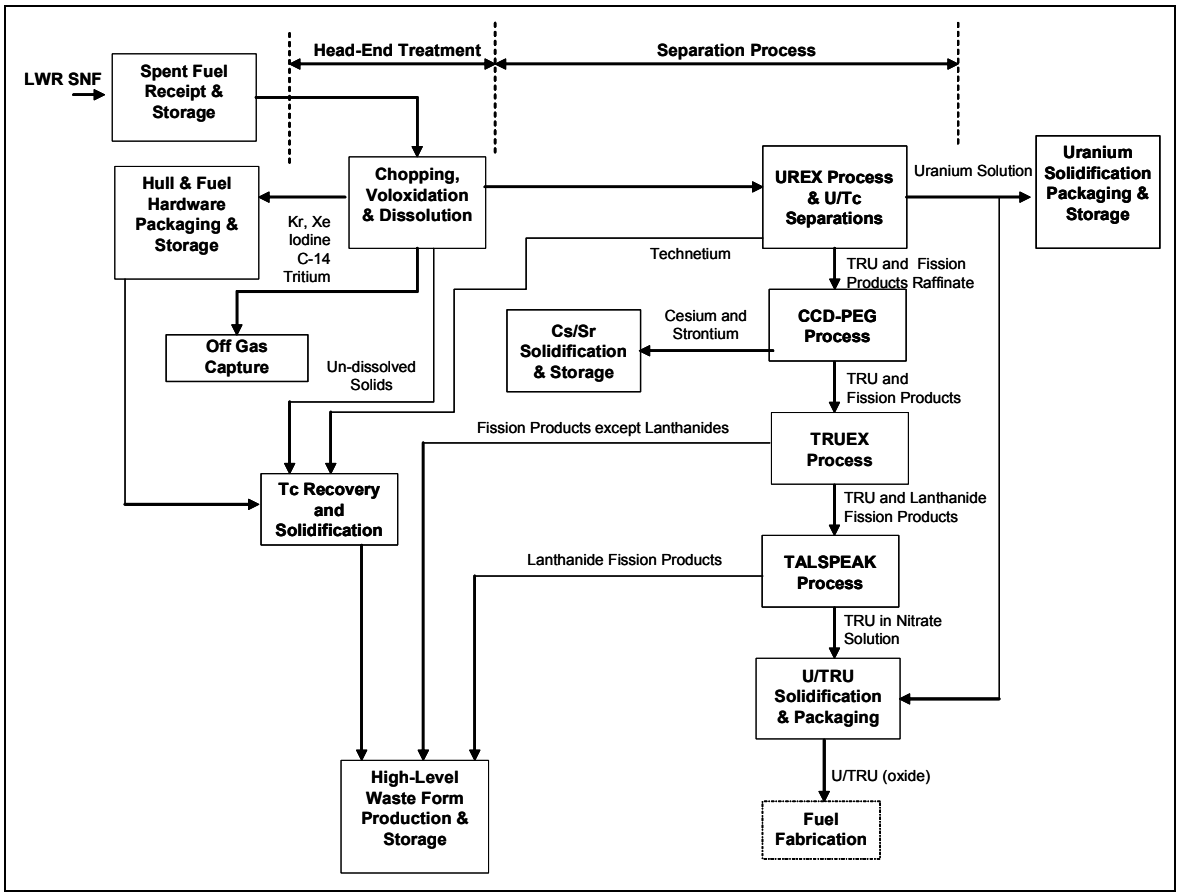
Figure A.5.1-3 presents a block flow diagram of the basic UREX+ aqueous reprocessing technology. As can be seen from that figure, LWR SNF is chopped and then dissolved in nitric acid. The solution from the dissolver is clarified to remove any particulate material and then it goes into the first solvent extraction process which is called UREX. The process uses tributyl phosphate, the same extractant used in PUREX, but by adding acetohydroxamic acid, it does not extract plutonium. This process also does a very efficient job of extracting the technetium along with the uranium.

Next, the technetium is separated from the uranium using an ion exchange column. The next step is to extract the technetium from the resin, convert it to a zirconium alloy and combine it with the cladding hulls and the sludge from the dissolver to produce a metallic waste form. This is done in order to place technetium in a large mass of zirconium, thus it would remain in the metallic state rather than in the oxide state. The technetium, which has a relatively low solubility in the metal state, would then be less likely to migrate to groundwater and would be less mobile in a geologic repository. This could reduce potential dose-related issues for technetium with respect to a geologic repository with an oxidizing environment.

The uranium extracted in the UREX process is converted to an oxide and stored in a highly purified state. This uranium can be stored without any shielding requirement for radiation protection.

The next step is to extract the cesium (Cs) and strontium (Sr) (along with rubidium [Rb] and barium [Ba]) from the UREX process waste stream. Multiple alternatives for disposing of a separate Cs/Sr waste stream might be considered. Under one alternative, the Cs and Sr would be stabilized and placed into decay storage for up to 300 years.

Next, the raffinate from the Cs/Sr separation process is transferred to the Transuranic Extraction Process (TRUEX). TRUEX is a well-developed process that is used in commercial applications. The TRUEX process extracts transuranic elements and lanthanides (rare earth fission products).



Source: WSRC 2008a
Note: Kr – Krypton; Xe – Xenon; C-14 – Carbon-14; UREX – Uranium Reduction and Extraction; Tc – Technetium; Cs – Cesium; Sr – Strontium; CCD-PEG - Chlorinated Cobalt Dicarbollide/Polyethylene Glycol; TRUEX - Transuranic Extraction; TALSPEAK - Trivalent Actinide Lanthanide Separations by Phosphorusreagent Extraction from Aqueous Complexes; U – Uranium; TRU – Transuranics

FIGURE A.5.1-3—*Uranium Reduction and Extraction Process Aqueous Reprocessing Technology Flow Diagram*

The TRU/lanthanide product from the TRUEX process is then transferred to the Trivalent Actinide Lanthanide Separations by Phosphorusreagent Extraction from Aqueous Complexes (TALSPEAK) process which is used to separate lanthanides from the transuranics. The lanthanides become part of the HLW stream. The transuranics then go to a step in which part of the uranyl nitrate solution from the UREX process is blended with the aqueous transuranic stream from the TALSPEAK process, which is then sent to the fuel conversion process where the liquid stream is converted to oxides. If the recycled fuel becomes an oxide, then the separation process is complete. If the fuel becomes metallic, then the oxides have to be reduced to metals. Finally, the oxides or metal go into fuel fabrication.

Aqueous Separations Waste Processing

There are three main waste streams generated from the UREX process: technetium (Tc), Cs/Sr, and fission products (including lanthanides). These are discussed below.

Technetium Recovery and Immobilization: As previously discussed, recovered metallic technetium would be alloyed with a portion of the fuel hulls and hardware to produce a metallic

waste form that would immobilize the technetium. This HLW form would be packaged for on-site storage (assumed up to the end of facility life) awaiting shipment to a geologic repository for disposal.

Cs/Sr Solidification: The Cs/Sr solution would be evaporated and subsequently solidified. The current baseline process would be to stabilize the components with additives to produce a solid waste form. One option is to store this waste for up to 300 years (at the site of the facility or a different location) to allow sufficient decay to reduce radiation and thermal output, then the Cs/Sr solid waste form would be disposed of in an appropriate facility. Another option for Cs/Sr would keep it with the fission product/lanthanide stream and make it into a borosilicate glass suitable for disposal in a high-level waste geologic repository.

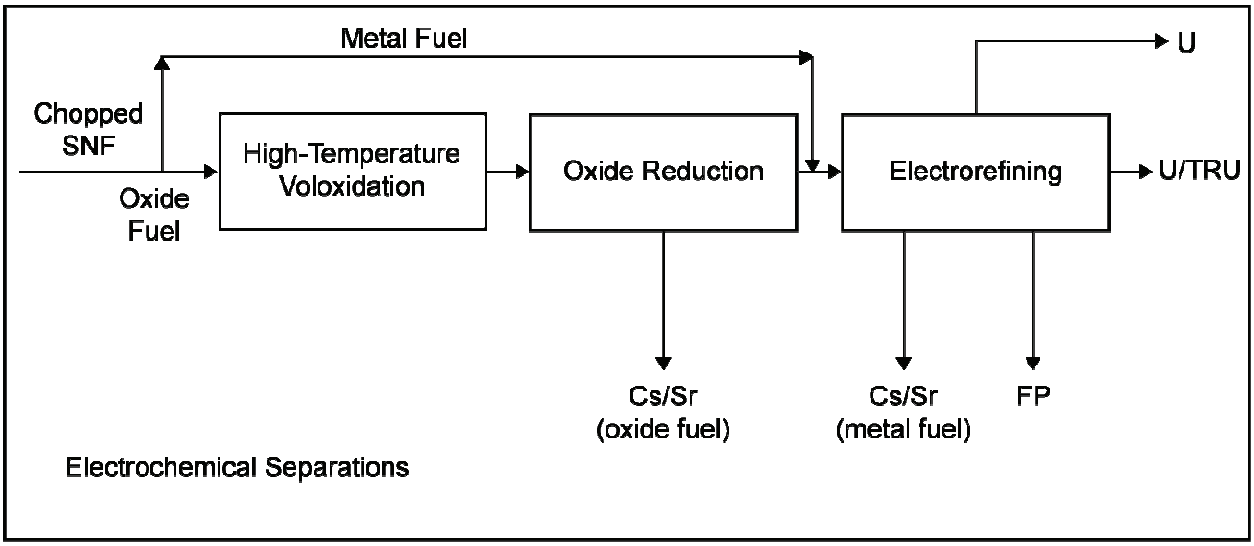
Fission-Product/Lanthanide Solidification: The fission product waste streams from TRUEX and lanthanides from TALSPEAK separations processes must be treated to a solidified, leach-resistant waste form suitable for disposal in a HLW geologic repository. Storage and cooling of the solidified HLW package would be required prior to shipment to the geologic repository (WSRC 2007d).

The UREX+ suite of processes offer several options for dealing with the transuranics. Because of their high radioactivity and decay heat, some of the minor actinides may not prove suitable for transmutation fuels and may be placed in the waste streams with the fission products. Development of stable waste forms for transuranic bearing wastes is being evaluated in the AFCI R&D program.

A.5.2 Electrochemical Separations

Electrochemical separations employs a molten salt electrorefiner to treat SNF. Electrochemical processing extracts two products, uranium by itself and uranium with all the transuranic elements together. It also produces several waste streams. Electrochemical processing has been in use for many years for purification of materials, including plutonium (Avens and Eller 2000). Electrochemical processing is a candidate process for recycling both fast reactor oxide and metal SNF and LWR oxide SNF. It was selected as the preferred method to treat 27.5 tons (25 MT) of DOE sodium-bonded metal fuel from Experimental Breeder Reactor-II (EBR-II) and Fast Flux Test Facility (FFTF) (DOE 2000e). Development tests for its application to LWR oxide fuel have been conducted at ANL and Idaho National Laboratory (INL) under the AFCI program, as well as DOE programs that preceded AFCI.

Figure A.5.2-1 presents a simplified block flow diagram of the basic process of electrochemical processing technology. This figure depicts an electrochemical process for both oxide and metal based fuel. The process is simpler for a metal based SNF because voloxidation and electroreduction would not be needed. The individual steps in the electrochemical treatment process are described below.



Source: WGI 2008a
Note: FP = fission products

FIGURE A.5.2-1—*Simplified Schematic Block Flow Diagram of Electrochemical Separations*

Disassembly and Fuel Element Chopping: The fuel elements that contain uranium and fission products are separated from the assembly hardware by cutting the assemblies and physically separating the fuel elements. The fuel elements are cut into small pieces and transferred to the voloxidation step. The assembly hardware is packaged for storage and disposal. This metal waste is remotely handled due to presence of highly radioactive activation products.

Voloxidation: To process oxide fuel through the electrorefiner, it must first be converted to metal using a combination of voloxidation and electrolytic reduction. Voloxidation, a process where the fuel is exposed to oxygen at elevated temperatures, converts UO_2 to triuranium octaoxide (U_3O_8), causing it to expand, break into small pieces, and separate from the cladding. Gaseous and volatile fission products are released to an off-gas capture system during this process. The fuel pieces are placed in porous metal baskets that become cathodes (negative electrodes where reduction of oxides to metals occurs) in the electroreduction step.

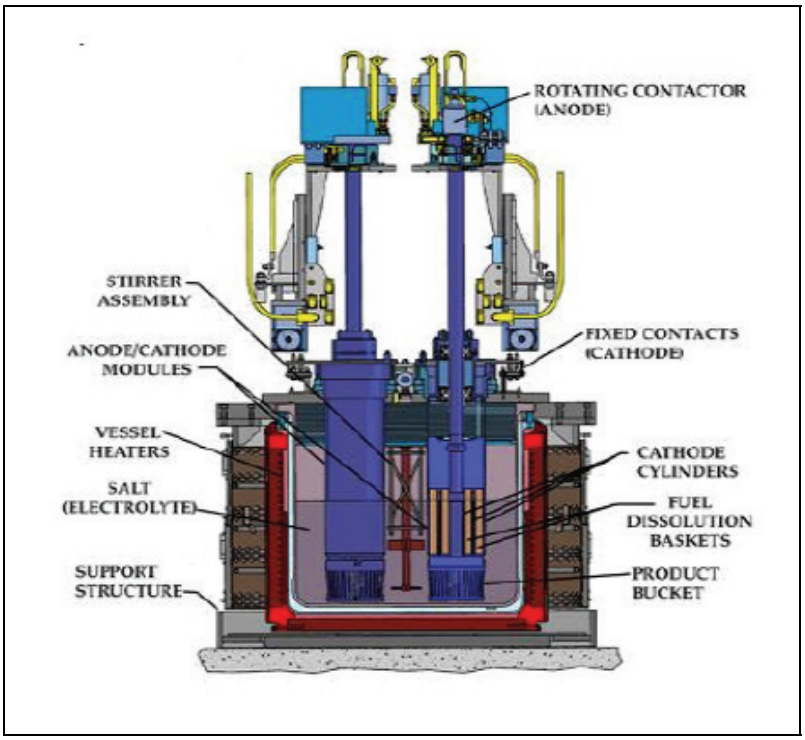
Electrolytic Reduction: Both metallic and oxide fuel could be treated in the electrorefining process. For oxide fuel, the separated oxide fuel particles must first be converted to metal by electrolytic reduction in a molten lithium chloride (LiCl) salt, releasing oxygen in the process. Cs, Sr, Ba, and Rb form chlorides and accumulate in the salt. Periodically they are removed via zeolite ion exchange and the cleaned salt is reused. The zeolite containing the Cs, Sr, Ba, and Rb is converted to a stable waste form and packaged for disposal. Gaseous and volatile fission products are released to an off-gas capture system during this process. The now metallic SNF is treated in the electrorefining process.

Electrorefining: The electrorefiner is where the main separation processes occur (Figure A.5.2-2). An electrorefiner is a heated steel vessel that contains a molten mixture of salts, primarily lithium chloride, potassium chloride, and uranium trichloride. It has two or more electrodes: one or more anodes (positive electrodes where oxidation occurs) and one or more

cathodes (negative electrodes where reduction occurs). The anodes are the baskets from electroreduction that hold the reduced SNF pieces, and each cathode consists of bare steel rods, where uranium metal is collected.

Upon application of an electric current between the anodes and cathodes, uranium is oxidized to ions at the anode and uranium ions are reduced and deposited as metal at the cathode. Plutonium, other transuranic elements, and the remaining active metal fission products are oxidized into the salt by chemical reactions that deplete the uranium chloride concentration by reducing uranium ions to metal at the fuel baskets. Solid dendritic uranium is deposited at the cathode until all the uranium in the anode baskets is oxidized. Then the uranium cathode product is raised into the gas space in the electrorefiner to allow molten salt to drain, although some salt adheres to the product. The cathode product is then removed from the electrorefiner. Uranium trichloride, produced by chlorinating some of the uranium product, is periodically added to the electrorefiner to replace the uranium reduced by the chemical reactions involving fission products and transuranics.

The cladding hulls and noble fission products remain undissolved in the anode baskets. They are removed from the electrorefiner and melted into waste-form ingots. The reactive fission products and transuranic elements accumulate in the electrorefiner salt. Additional electrorefining or electrolysis steps separate the transuranics from the reactive fission products.



Source: Vaden et al. 2007

FIGURE A.5.2-2—*Diagram of Electrorefiner Used to Treat Experimental Breeder Reactor-II Fuel at INL*

U/TRU Recovery: An electrolysis or electrorefining process is used to recover the uranium and transuranics as a mixed uranium/transuranic (U/TRU) metal product with some lanthanides.

Following U/TRU recovery the salt is further processed to remove the remaining lanthanides and other fission products. The salts are recycled back to the electrorefiner and the fission product waste is converted to a glass-bonded ceramic for disposal.

Cathode Processing: The uranium and uranium-transuranic products are removed from the electrorefiner and treated to remove any adhering salt in the cathode processor, which is a specially designed retort furnace equipped with a vacuum system. The cathode product is heated under vacuum, separating the salt by distillation and melting the remaining metal into an ingot. The salt is recycled back to the electrorefiner.

In summary, electrochemical separations would produce two products and generate five waste streams from the processing of SNF. The two products would be a uranium metal and a U/TRU metal, which would be used for fabrication of fuel for the advanced recycling reactor. These metals can be converted to oxide and stored or kept in metal form. The five waste streams would be assembly hardware, fission-product off-gases, Cs/Sr salt, cladding metal, and fission product salt wastes. The assembly hardware and off-gases would be stabilized and packaged for disposal using established radioactive waste treatment methods. The Cs/Sr combined with the zeolite would be converted to a glass-like form and disposed of in a geologic repository or stored until it had sufficiently decayed for disposal at a LLW facility. The metal wastes would be converted to ingots and packaged for onsite storage awaiting shipment to a geologic repository for disposal. The fission product waste would be solidified in a glass matrix, packaged, and stored onsite awaiting shipment to a geologic repository.

A.5.3 Oxidation and Reduction of Oxide Fuel (OREOX) Process

The Thermal Reactor Recycle Alternative (Option 2) could utilize a dry recycling technology. This technology includes mechanical removal of the cladding followed by a thermal process to reduce the spent LWR fuel to powder, which is then sintered, and pressed into CANDU-sized pellets. This separation and fuel fabrication process has been termed the Oxidation and Reduction of Oxide Fuel (OREOX) process (Yang et. al. 2005). Because the OREOX process is part of a South Korean program, in a research stage, only open literature publications are available. Consequently, there is less information available for this fuel separation technology than many of the other technologies presented in Section A.5. However, because the OREOX process only uses mechanical and thermal processes (which are similar to the front end step used in both the aqueous and electrochemical separation processes), it is expected that the environmental impacts (e.g., emissions, radiation dose to workers, and wastes) would not be greater than those from other processes described above. While the OREOX process employs mechanical processes and a detailed process design must still be developed, the potential exists for the introduction of liquids for decontamination, process cleanup, or similar secondary operations. In the event liquids are used in this process and small quantities of liquid HLW are generated, appropriate treatment processes would be included to solidify these wastes such that no liquid HLW would be accumulated or require long term storage. Fuel fabrication is addressed in the following paragraphs.

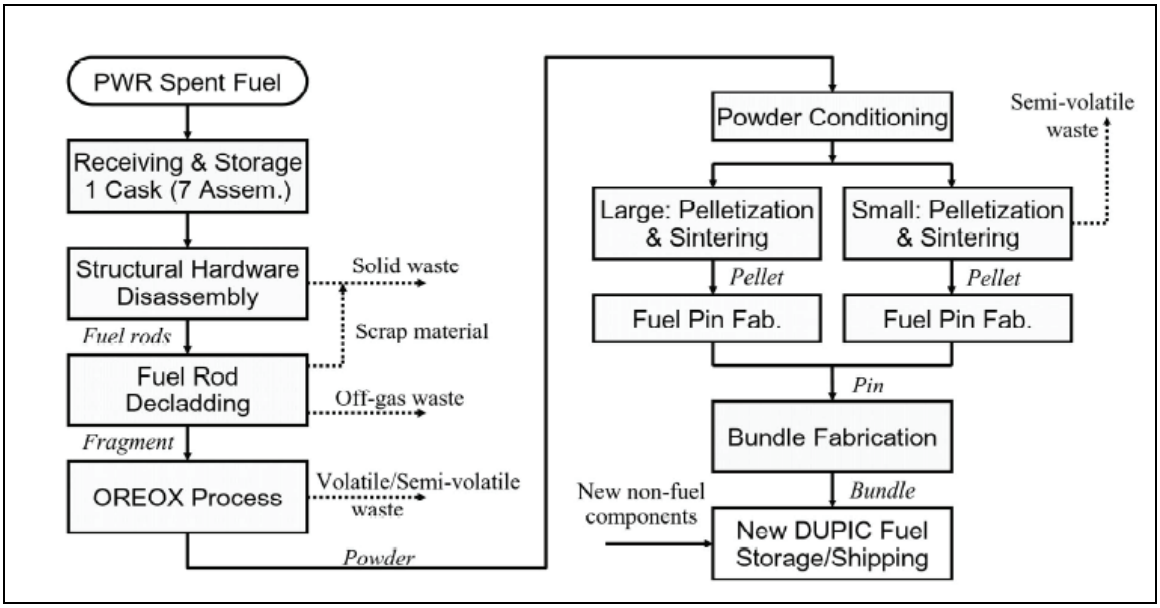
The Korea Atomic Energy Research Institute (KAERI) has fabricated DUPIC fuel elements in a laboratory-scale remote fuel fabrication facility. KAERI has demonstrated the fuel performance

in a research reactor, and has confirmed the operational feasibility and safety of a CANDU reactor loaded with the DUPIC fuel using conventional design and analysis tools, which will be the foundation of the future practical and commercial uses of DUPIC fuel (Yang et al. 2005). Figure A.5.3-1 depicts the DUPIC fuel fabrication process using OREOX.

A DUPIC Fuel Fabrication Facility would consist of the following five processes:

PWR Spent Nuclear Fuel Receiving and Storage: The LWR SNF would be transported to the DUPIC facility in a standard LWR SNF transport cask.

Spent Nuclear Fuel Disassembly and Decladding: The LWR SNF rods would be mechanically removed from the LWR fuel assembly. The fuel assembly structural material would be transferred to the solid waste treatment area and stored after going through the volume reduction and packaging process. The fuel rods would be punctured, chopped, or cut (depending on the specific technique selected) into an appropriate size by a mechanical and/or laser cutting method, and the oxide fuel material and cladding would be mechanically separated. The fission gases released in this process would be sent to the off-gas treatment system and would be stored. The cladding material would be cleaned and decontaminated for more than a 99 percent recovery of the fuel material.



Source: Yang et al. 2005

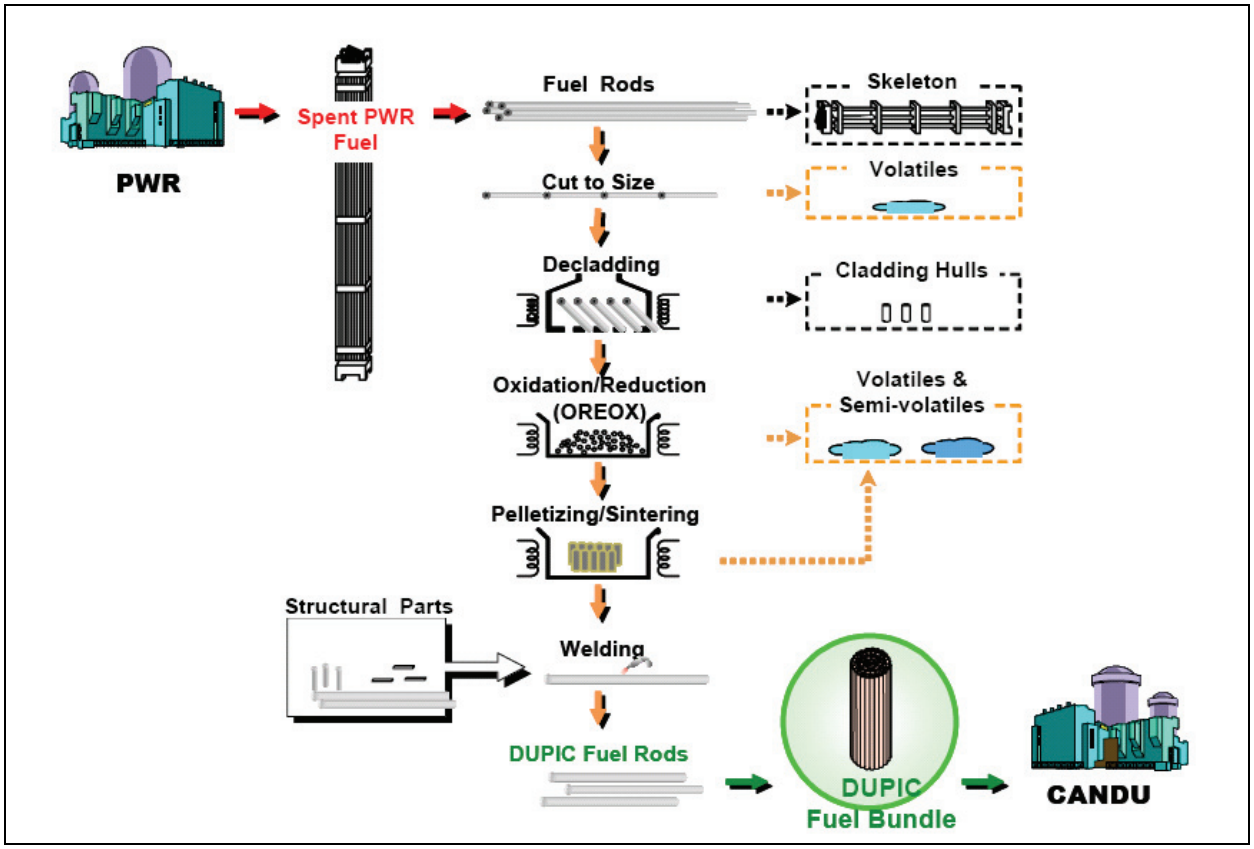
FIGURE A.5.3-1—*Direct Use of Spent Pressurized Water Reactor Fuel in Canada Deuterium Uranium Fuel Fabrication Process Using Oxidation and Reduction of Oxide Fuel*

Fuel Powder Preparation: The LWR SNF materials would be treated by the OREOX process to form fuel powder that satisfies the powder characteristics requirements. The OREOX process consists of multiple (typically three) cycles of oxidation at 930°F (500°C) with air followed by reduction at 1,300°F (700°C) with hydrogen. OREOX processing removes the bulk (greater than 85 percent) of the volatile fission products (tritium, iodine, krypton/xenon, and carbon-14 [C-14]). The semi-volatiles (Cs and transition metals) are less affected by OREOX but are still

partially (less than 50 percent) removed. A disposition path for these fission products would be required for their disposal/storage. Figure A.5.3-2 depicts the process of fabricating DUPIC fuel.

Fuel Pellet Fabrication: The DUPIC fuel pellets would be produced from the LWR SNF powder through the pre-compaction, granulation, final compaction, sintering, and the grinding processes.

Fuel Element Fabrication: The fuel pellets would be loaded into the cladding tube manufactured outside the hot cell and the end cap would be welded to form a fuel element. These fuel elements would then be bundled into fuel bundles and the fresh DUPIC fuel would be transported to a CANDU reactor. Although a fraction of the gamma radioactivity would be removed from the recycled fuel, gamma radioactivity would still be high enough to require all refabrication and handling to be done remotely in a shielded facility (Yang et al. 2005).



Source: Yang and Park 2006

FIGURE A.5.3-2—*Direct Use of Spent Pressurized Water Reactors Fuel in Canada Deuterium Uranium Fuel Fabrication Process*

A.6 NUCLEAR FUEL RECYCLING CENTERS

Nuclear fuel recycling centers associated with the Fast Recycle Alternative, Thermal/Fast Recycle Alternative, and Thermal Recycle Alternative (options 1 and 3) could be comprised of as many as three individual components: 1) a LWR SNF separations facility; 2) a fuel fabrication facility for transmutation fuel and/or MOX-U-Pu fuel[14]; and 3) a separations facility for SNF from a fast reactor and/or MOX-U-Pu LWR[15]. These components would not necessarily all be in the same location. A nuclear fuel recycling center could be operated as a large centralized facility or smaller distributed facilities. Examples for each of the component facilities are described below.

A.6.1 Light Water Reactor Spent Nuclear Fuel Recycling Facility

A LWR SNF recycling facility would separate the SNF discharged from LWRs into its reusable components and waste components. Depending on the programmatic alternative and technology, this could include:

- Separation of high purity uranium from the SNF that would allow recycle for re-enrichment or for other use or disposition.
- Separation and immobilization of long-lived fission products, including technetium and iodine, for disposal in a geologic repository.
- Potential extraction and storage (up to 300 years) of short-lived fission products (cesium and strontium).
- Separation of transuranic elements without separating pure plutonium for fabrication into fuel for an advanced recycling reactor and/or thermal-spectrum reactor (WSRC 2008a).

The LWR SNF recycling facility would be a self-sufficient operation located at a greenfield site or an existing facility. The following throughput options are analyzed for the nuclear fuel recycling center as a greenfield site: a lower bound of 100 MTHM per year and a base case of 800 MTHM per year. Selection of 800 MTHM per year as the base case was premised on existing separations facilities overseas. The 100 MTHM could be a demonstration size facility. This separations capacity is the basis for establishing the impacts at the production levels of 100 GWe, 150 GWe, 200 GWe, or 400 GWe.

A.6.1.1 *Separations Process Description*

The LWR SNF recycling facility would receive and manage SNF and use a separations process designed to recover the desired materials such as the examples discussed in the previous section. Key facility operations for recycling of LWR SNF include:

[14] This section presents information related to fuel fabrication facility for transmutation fuel. Fuel fabrication of MOX-U-Pu fuel is expected to be similar in terms of facility requirements. See Section A.3.1.4 for a discussion of MOX-U-Pu fuel fabrication.

[15] This section presents information related to separation of fast reactor SNF, which is representative of the facility requirements that would be expected for separating MOX-U-Pu SNF.

- SNF receipt, storage, and transfer
- SNF preparation and head-end treatment
- Separations processing and purification
- Product handling—solidification, packaging, storage, and shipping of uranium and U/TRU oxides
- Waste processing and handling–packaging, storage and preparation for shipment of wastes

The base case alternative allows for a maximum of 800 MTHM of PWR fuel to be recycled each year. This would equate to approximately 1,820 SNF assemblies per year. The wet and dry storage areas would each have the capability to store approximately 1,820 SNF assemblies. The facility baseline would be able to store 2 years throughput in the combined wet and dry storage areas (WSRC 2008a). Table A.6.1.1-1 provides the maximum number of SNF assemblies to be processed annually under each scenario.

TABLE A.6.1.1-1—*Maximum Number of Spent Nuclear Fuel Assemblies Processed Annually*

	100 MTHM/year Facility	800 MTHM/year Facility
Annual Number of PWR [a] SNF Assemblies	228	1,820

Source: WSRC 2008a
[a] Only values for PWRs are being given since they are the more prominent of the LWR fuel types. Boiling water reactor assemblies are smaller than PWR assemblies and would fit into the same storage locations.

Details on construction requirements and the environmental impacts would be developed at the time a specific proposal was being made for design and construction of a LWR SNF recycling facility.

A.6.1.2 *Facility Requirements*

Based on preconceptual design studies for a reprocessing plant using aqueous separations technologies, the total site area for the LWR SNF recycling facility within a property protection fence would be on the order of 500 acres (202 hectares [ha]) for an 800 MTHM per year facility. The 100 MTHM per year facility layout would be similar but occupy less total acreage (approximately 300 acres [121 ha]). An aqueous reprocessing facility is used as the representative technology for processing LWR SNF since there is no experience with other processing technologies at large scale.

The footprint for the processing areas is estimated to be on the order of 1,000,000 square feet (ft^2) (92,900 square meters [m^2]) for the 800 MTHM facility. Table A.6.1.2-1 provides the footprint area for the LWR SNF recycling facility using aqueous separations technology (WSRC 2008a).

TABLE A.6.1.2-1—*Light Water Reactor Spent Nuclear Fuel Recycling Facility Building Footprint Based on Representative Technology*

	Area (ft^2)	
	100 MTHM/year Facility	**800 MTHM/year Facility**
Total Area of Main Processing Buildings	520,000	1,040, 000
Total Support Building Area	1,140, 000	2,280, 000
Total Building Area	1,660, 000	3,320, 000

Source: WSRC 2008a

A.6.1.3 *Operational Materials and Waste*

During normal operations, the LWR SNF recycling center would process SNF to recover uranium and transuranic products and produce waste materials. Throughputs and inventories for recovered uranium and transuranics products and waste materials are shown in Tables A.6.1.3-1 and A.6.1.3-2 (WSRC 2008a). Estimates of waste mass and volume, based on existing aqueous technologies and experience, are provided in Table A.6.1.3-3. Application of waste minimization, pollution prevention, and actual facility designs are expected to reduce these amounts. Additionally, the data for the LWR SNF recycling center were developed by assuming that the burnup of the LWR SNF would be 60 GWd/MTHM, which is slightly higher than expected (51 GWd/MTHM—see Table 4.8-1). As a result, these data are expected to be conservative in estimating environmental impacts because higher burnup LWR SNF would produce larger quantities of fission products in the spent fuel. Advanced technologies already exist, and others are the subject of R&D, that have the potential to reduce the mass or volume of some waste streams, but estimates for a facility that would use such technologies would be developed at the time a specific facility and processing approach was proposed. Table A.6.1.3-4 provides a summary of the operations data for the LWR SNF recycling facility.

TABLE A.6.1.3-1—*Estimates of Fuel Processing Materials and Wastes from Light Water Reactor Spent Nuclear Fuel Recycling Facility Operations 100 Metric Tons Heavy Metal/Year Facility*

Feed/Product/Waste	Annual Rate (kg)	Annual Bulk Container Rate
LWR Fuel Feed (fuel portion only)	113,073	228 assemblies
U Solidification and Storage Product	106,847	267 55-gallon drums
U/TRU Solidification and Storage Product	5,490	384 cans [a]
Fuel Hardware and Hulls Waste	41,218	12 canisters [b]
Tc Metal Alloy Waste Form	2,380	1 canisters [b]
Cs/Sr Waste Form	9,408	510 canisters [c]
FP/Lanthanide Vitrified Waste Form	38,649	13 canisters [d]

Source: WSRC 2008a
[a] Can holds 14.3 kg of material
[b] Canister holds 3,600 kg of material
[c] Canister holds 18.5 kg of material
[d] Canister holds 2,900 kg of material

TABLE A.6.1.3-2—*Estimates of Fuel Processing Materials and Wastes from Light Water Reactor Spent Nuclear Fuel Recycling Facility Operations 800 Metric Tons Heavy Metal/Year Facility*

Feed/Product/Waste	Annual Rate (kg)	Annual Bulk Container Rate
LWR Fuel Feed (fuel portion only)	903,850	1,820 assemblies
U Solidification and Storage Product	853,920	2,135 55-gallon drums
U/TRU Solidification and Storage Product	43,872	3,068 cans[a]
Fuel Hardware and Hulls Waste	329,410	92 canisters[b]
Tc Metal Alloy Waste Form	19,022	6 canisters[b]
Cs/Sr Waste Form	75,185	4,075 canisters[c]
FP/Lanthanide Vitrified Waste Form	308,880	106 canisters[d]

Source: WSRC 2008a
[a] Can holds 14.3 kg of material
[b] Canister holds 3,600 kg of material
[c] Canister holds 18.5 kg of material
[d] Canister holds 2,900 kg of material

TABLE A.6.1.3-3—*Estimates of Wastes from Light Water Reactor Spent Nuclear Fuel Recycling Facility Operations*

Waste Category	100 MTHM/year Facility	800 MTHM/year Facility
	Annual	Annual
LLW		
Liquid (L)	274	2,156
Solid (m^3)	4,043	7,936
Mixed LLW		
Solid (m^3)	11	32
GTCC LLW		
Solid (m^3)	707	1,250
Mixed GTCC LLW		
Solid (m^3)	10	77
HLW		
Solid (m^3)	28	221
Hazardous		
Liquid (L)	70	100
Solid (m^3)	32	93
Nonhazardous		
Liquid (L)	181,000,000	248,000,000
Solid (m^3)	11,328	16,463

Source: WSRC 2008a

TABLE A.6.1.3-4—*Summary of Light Water Reactor Spent Nuclear Fuel Recycling Facility Operations Data Based on Representative Technology*

Data Required	100 MTHM/year Facility	800 MTHM/year Facility
Electrical Consumption – daily and annual	3.0 GWh 906 GWh	6.0 GWh 1,440 GWh
Peak electrical demand (MVA) – daily	195	390
Diesel Fuel usage (gal) –annual	121,750	243,500
Other Process Gas (N, Ar, etc.) – daily and annual	4,689 scf/day 1,125,440 scf/year	14,068 scf/day 3,376,320 scf/year
Domestic Water (gal) – daily and annual	135,000 40,770,000	175,000 42,000,000
Process Water (gal) – daily and annual	120,000 36,240,000	360,000 108,720,000
Cooling Tower Makeup (gal) – daily and annual	249,600 59,904,000	748,800 179,712,000
Steam (gal) – daily and annual	307,200 73,728,000	921,600 221,184,000
Employment (total workers)	2,500	3,100
Number of Radiological Workers	1,070	2,226
Average annual dose to Radiological workers (mrem)	250	250
Maximum annual worker dose (mrem)	1,000	1,000

Source: WSRC 2008a
Note: scf=standard cubic feet

The operation of fuel cycle facilities generates several different types of waste. Some are closely related to the process and throughput (e.g., fission products, used solvents, product packages and containers, and excess acid). Other waste streams (secondary wastes) are more closely related to staffing (e.g., sanitary waste) or plant systems and facilities (filters, laboratory wastes, and decontamination material). However, the largest source of secondary radioactive waste is typically associated with routine operation and maintenance of the nuclear facilities and equipment. Estimates of total waste were derived by combining "process-related" wastes directly related to throughput, with estimates of secondary waste made for each facility. Estimates of secondary wastes streams considered process conditions, personnel activities (entries into contamination areas and protective clothing requirements), and forecasts of equipment failures, repairs, and replacement. Detailed estimates considering forecasts of routine operations and both major and minor maintenance activities were prepared for each case. Since the total quantity of waste for any given case is impacted by all of these factors, and their relative contribution varies with the type of operation and source materials, comparisons between cases are unlikely to be directly proportional to throughput except for process wastes. Detailed estimating methodology and facility specific assumptions are described in the document *Engineering Alternative Studies for Separations, Waste Generation Forecast and Characterization Study 800MT/Year UREX +1A*, (WSRC 2008e).

The current footprint for the 800 MTHM/year LWR SNF recycling facility assumes only 1 year of storage for each of the types of waste. To account for the potential for additional storage capacity, the footprint for the LWR SNF recycling facility would need to be increased by the following values depending on waste stream. It is expected, however, that a disposal pathway for these wastes may be available while the facility is operating. If so, additional storage capacity would not be required. A phased construction plan with expandable capacity is envisioned to

handle this waste and provide sufficient but not excess storage capacity. New capacity would be built every 5 to 10 years to accommodate a portion of the total waste that would be generated during the subsequent years of production. The need for the construction of new storage space would be reduced or eliminated when disposal paths for the various waste categories are decided. The following values represent the storage capacity required per year for wastes generated from the 800 MTHM/year LWR SNF Recycling Facility.

- 3,260 ft^2/yr (300 m^2/yr) for HLW storage (includes hulls and hardware)
- 8,150 ft^2/yr (760 m^2/yr) for Cs/Sr waste storage
- 13,600 ft^2/yr (1,300 m^2/yr) for Greater-than-Class C (GTCC) LLW storage
- 111,260 ft^2/yr (10,340 m^2/yr) for LLW storage (includes any grouted LLW)
- 2,300 ft^2/yr (200 m^2/yr) for combined hazardous waste/mixed waste storage

Hulls and hardware would be GTCC LLW, but due to the need for remote handling they are assumed be placed in the same storage facility as the HLW to reduce the need for two storage facilities with remote handling capabilities (WSRC 2008a).

A.6.2 Transmutation Fuel Fabrication Facility

A.6.2.1 *Process Description*

The transmutation fuel fabrication facility would receive uranium and U/TRU product from one or more nuclear fuel recycling centers. The assumption has been made that the fuel for the advanced recycling reactors is a ceramic oxide to provide an example for evaluating environmental impacts in this GNEP PEIS. Other transmutation fuel types are available, such as metal, nitride, and carbide, and the selection of the fast reactor fuel type would be made as part of the development of the advanced recycling reactor, as discussed in the following section. Options also include using more than one composition for the transmutation fuel, such as U/Pu for some of the fast reactor fuel, and retaining the minor actinides in other fast reactor "target" assemblies. Key facility operations for fuel fabrication are the same for different fuel types. The differences are in the development of the feedstock prior to arriving at the fuel fabrication facility.

Key facility operations for fabrication of transmutation fuel include:

- Oxide product receipt, storage, and transfer
- Conditioning and fabrication
  - Dissolving, blending, and solidifying feedstock in an oxide form suitable for fuel fabrication
  - Stoichiometry adjustment, mixing, milling, and binder addition
  - Wet and dry scrap processing
  - Pressing, sintering, and grinding of fuel pellets
  - Fuel rod loading and fuel bundle assembly
- Fuel assembly, handling, and storage
- Waste processing and handling–packaging, storage, and preparation for shipment of wastes (WSRC 2008b)

The United States currently has three NRC-licensed uranium fuel fabrication facilities capable of processing UF_6 to UO_2 powder and then fabricating LWR fuel assemblies from this UO_2 powder. Three additional facilities, Nuclear Fuel Services, in Erwin, TN, BWX Technologies, in Lynchburg, VA, and Areva NP, in Lynchburg VA, are NRC licensed, but currently do not have the ability to process UF_6 to UO_2 powder. Table A.6.2-1 shows the capacity of the three facilities presently able to produce commercial LWR fuel assemblies. The current LWRs require approximately 2,400 tons (2,170 MT) of fresh fuel assemblies annually.

TABLE A.6.2-1—*United States Light Water Reactor Fuel Fabrication Capacity*

Facility	Location	Capacity (Metric Tons)
Global Nuclear Fuel-Americas, LLC	Wilmington, NC	1,200
Westinghouse	Columbia, SC	1,600
Areva NP, Inc.	Richland, WA	700
TOTAL		**3,500**

Source: NRC 2007c

For fast reactor fuel fabrication, a size of 100 MTHM/year was used, assuming one centralized facility rather than a number of smaller distributed fuel fabrication facilities colocated with the fast reactors. A 100 MTHM/year capacity will provide fast reactor fuel for 11 GWe of fast reactor capacity (see Section 4.3.1). The assumed inventory of nuclear material contained in the various processes consists of approximately 1 year of U and TRU feedstock storage and 2 years production of fuel assembly storage (WSRC 2008b).

A.6.2.2 *Facility Requirements*

The transmutation fuel fabrication facility includes process buildings and support buildings. The total site area within a property protection fence would be on the order of 100 acres (41 ha) for the 100 MTHM/year transmutation fuel fabrication facility. The proposed concept places the fuel fabrication into one main building. However, this building would be separated into several parts and cells to accommodate the various processes. The storage areas may be in separate buildings, especially for the fuel assemblies (WSRC 2008b).

The footprint for the processing areas is estimated to be on the order of 520,000 ft^2 (48,300 m^2) for the 100 MTHM/year facility. The process area footprint provides space for processing area support functions and would include various tunnels for the transfer of materials between buildings and other colocated facilities (WSRC 2008b). Table A.6.2.2-1 provides the footprint area discussed above.

TABLE A.6.2.2-1—*Transmutation Fuel Fabrication Facility Building Size Details*

	Area (ft^2)
Total Area of Main Processing Buildings	520,000
Total Support Building Area	1,840,000
Total Building Area	2,360,000

Source: WSRC 2008b

Details of the construction requirements and the environmental impacts would be developed at the time a specific proposal was made for design and construction of a transmutation fuel fabrication facility.

A.6.2.3 *Operational Materials and Waste*

Throughputs and inventories of processing materials, shown in Table A.6.2.3-1, are based on the conceptual process flow sheets that are currently under development in the AFCI program (WSRC 2008b). Process storage requirements would approximately be one year for the U/TRU oxide feed stock, bulk fuel materials, and undissolved solids, with two years of product storage for completed fuel assemblies. Estimates of all the operations wastes, including process wastes, are provided in Table A.6.2.3-2 to the extent available.

TABLE A.6.2.3-1—*Estimates of Fuel Processing Materials and Wastes from Transmutation Fuel Fabrication Operations 100 Metric Tons Heavy Metal/Year Facility*

Feed/Product/Waste	Annual Rate (kg/year)	Annual Bulk Container Rate
U/TRU Oxide Feed	117,600	See Note
Bulk Fuel (Ceramic Oxide)	116,400	NA [a]
Undissolved Solids	120	NA [a]
Fuel Assemblies (# of assemblies)	1,680	1,680 assemblies

Source: WSRC 2008b
[a] In process stream
Note: Uranium oxide would be stored in 55-gallon drums (400 kg per drum) and U/TRU would be stored in containers that hold up to 14.8 kg of material.

TABLE A.6.2.3-2—*Estimates of Wastes from Transmutation Fuel Fabrication Operations*

Waste Category	Annual Volume 100 MTHM/year Facility
Low Level	
Liquid (L)	1,000
Solid (m^3)	2,367
Mixed Low-level	
Solid (m^3)	18
GTCC LLW	
Solid (m^3)	500
Hazardous	
Liquid (L)	33
Solid (m^3)	14.3
Nonhazardous	
Liquid (L)	55,300,000
Solid (m^3)	19,500

Source: WSRC 2008b

A.6.3 Fast Reactor Spent Nuclear Fuel Recycling Facility

A.6.3.1 *Process Description*

A fast reactor SNF recycling facility would receive and manage fast reactor SNF, using one or more processing technologies to separate the various components of SNF. Consistent with the

assumption for the transmutation fuel fabrication facility, the assumption has been made that the SNF from the advanced recycling reactor is a ceramic oxide. For the purpose of estimating environmental impacts from this facility, it has also been assumed that an aqueous separations technology (UREX+1a) is used, since there is no experience with other processing technologies at large scale and, for the purposes of this GNEP PEIS, this example provides an estimate of the potential environmental impacts from such a facility. If such facilities are eventually proposed, the detailed environmental impacts including construction data would need to be evaluated at that time.

Key facility operations for recycling of fast reactor SNF include:

- SNF receipt, storage, and transfer
- SNF preparation and head-end treatment
- Separations processing and purification
- Product handling—solidification, packaging, storage, and shipping of uranium and U/TRU oxides
- Waste processing and handling—packaging, storage, and preparation for shipment of wastes (WSRC 2008c)

The baseline process throughputs for 100 MTHM/year are calculated using a maximum 0.42 MTHM rate of fast reactor SNF per day (see Section A.6.1 for a description of the throughput rates for LWR and Fast Reactor separations.). These process throughputs were used to develop baseline equipment designs and layouts, which, in turn, were used to develop a theoretical plan for the entire facility. The inventory of nuclear material contained in the various separations and storage processes is presented in Table A.6.3.1-1 for the 100 MTHM/year facility, consistent with the fast reactor transmutation fuel fabrication facility (WSRC 2008c).

TABLE A.6.3.1-1—*Fast Reactor Spent Nuclear Fuel Recycling Facility Inventory of Nuclear Materials for Defining the Operations Basis*

Facility Description	100 MTHM/year Facility
Process Area	Annual material processing throughput 250 GWD/MTHM, 1 year cooled, fast reactor SNF
SNF Storage	At the baseline rate of 100 MTHM/yr, the 2-year storage capacity equates to receipt of 2,400 fast reactor SNF assemblies. Isolate and manage a minimum of 5 percent fuel assemblies received that may be damaged or otherwise unsuitable for near-term processing.

Source: WSRC 2008c

A.6.3.2 *Facility Requirements*

The fast reactor SNF recycling facility includes process buildings and support buildings. The total site area within a property protection fence is on the order of 250 acres (101 ha) (WSRC 2008c). The proposed concept would place most of the processes into as few buildings as possible.

The footprint for the processing areas is estimated to be on the order of 624,000 ft^2 (58,000 m^2) for the 100 MTHM/year facility. The process area footprint provides space for processing area support functions. In the current concept, the shielded areas are placed below grade (ranging from 20 to 40 ft [6 to12 m]), and the overhead cranes and other support equipment required for unloading and moving shipping casks and processing equipment extend to heights averaging 70 ft (21 m) above grade. Some buildings may require building heights greater than 70 ft (21 m) above grade. The process areas also include various tunnels for the transfer of materials between buildings (WSRC 2008c). Table A.6.3.2-1 provides the footprint area discussed above.

TABLE A.6.3.2-1—*Fast Reactor Spent Nuclear Fuel Recycling Facility Building Size Details*

	Area (ft^2)
	100 MTHM/year Facility
Total Area of Main Processing Buildings	620,000
Total Support Building Area	610,000
Total Building Area	1,230,000

Source: WSRC 2008c

Details of the construction requirements and the environmental impacts would be developed at the time a specific proposal was being made for design and construction of the fast reactor SNF recycling facility.

A.6.3.3 *Operational Materials and Waste*

During normal operations, the fast reactor SNF recycling facility would process SNF to produce uranium and transuranic products and waste materials. Estimated throughputs and inventories of these processing materials, shown in Table A.6.3.3-1, are based on the conceptual process flow sheets. Estimates of the operations data are provided in Table A6.3.3-2 to the extent available (WSRC 2008c).

TABLE A.6.3.3-1—*Estimates of Fuel Processing Materials and Wastes from Fast Reactor Spent Nuclear Fuel Separations Operations (100 Metric Tons Heavy Metal/Year Facility)*

Feed/Product/Waste	Annual Rate (kg)	Annual Bulk Container Rate
Transmutation SNF (assemblies)	223,000	1,200 assemblies
U Solidification and Storage Product	30,000	75
TRU Solidification and Storage Product	55,900	3,910
Fuel Hardware and Hulls Waste	69,700	20
Tc Metal Alloy Waste Form	36,900	11
Cs/Sr Waste Form	24,800	1,340
FP/Lanthanide Vitrified Waste Form	414,000	143

Source: WSRC 2008c

TABLE A.6.3.3-2—*Estimates of Wastes from Fast Reactor Spent Nuclear Fuel Separations Operations*

Waste Category	Annual Volume 100 MTHM/year Facility
	Annual
Low Level	
Liquid (L)	340
Solid (m^3)	5,050
Mixed Low-level	
Solid (m^3)	50
GTCC LLW	
Solid (m^3)	880
Mixed GTCC LLW	
Solid (m^3)	10
HLW	
Solid (m^3)	220
Hazardous	
Liquid (L)	88
Solid (m^3)	40
Nonhazardous	
Liquid (L)	223,000,000
Solid (m^3)	17,200

Source: WSRC 2008a

A.7 FUEL BURNUP

This PEIS includes a number of fuel cycle alternatives. These alternatives range from the current fuel cycle to fuel cycles which have only been studied on paper. This has resulted in varying levels of optimism based on the data used for the different fuel cycles. One factor in particular is the burnup assumed for each type of fuel at the time of discharge from the reactor. Burnup refers to the amount of energy generated per initial mass of fuel, the metric tons of initial heavy metal (MTIHM). For fuel assemblies of equal initial mass and for a given total energy production, higher burnup fuels can reduce the total mass of SNF generated by providing more energy per fuel assembly. However, each assembly would also contain a greater inventory of fission products for all enriched uranium-based fuels[16], and a greater inventory of heavier actinide elements. This section assesses and compares different discharge burnups and discusses the impact on a number of metrics, including SNF quantities generated, transportation required, raw materials utilization, and waste parameters (long-term heat and long-term radiotoxicity). The source of information in this section is "The Impact of Burnup on the Performance of Alternative Fuel Cycles" (Dixon and Wigeland 2008).

Table A.7-1 shows all of the alternatives that were analyzed in detail in the GNEP PEIS. Other alternatives were also discussed briefly in the text of the PEIS, such as the "deep burn" alternative for the HTGR. Since the neutron spectrum affects the probability of neutron capture (absorption) or fission, which in turn will affect the composition of the discharged fuel, rather

[16] This includes the specific fuel in the Thorium Alternative, which uses enriched uranium as a driver in both the seed and blanket. Thorium fuels can also use plutonium as a driver.

than a discussion of each reactor type, the reactors can be grouped by their neutron spectrum characteristics to determine the general impact of burnup.

- Thermal spectrum—No Action, Thorium, HWR, HTGR, LWR portion of all recycle alternatives and HWR portion of DUPIC alternative
- Fast spectrum—Advanced recycling reactor portion of Fast and Thermal/Fast recycle alternatives

TABLE A.7-1—*Programmatic Environmental Impact Statement Alternatives Indicating Assumed Reactor Mixes and Fuel Burnup Levels*

Case Description	No Action (Once-Through Fuel Cycle)	HWR or HTGR Alternative (Once-Through Fuel Cycle)		Thorium Alternative (Once-Through Fuel Cycle)	Thermal Recycle Alternative		Fast Recycle Alternative (CR=0.5)	Thermal /Fast Recycle Alternative (CR=0.5)
		All HWR	All HTGR		Option 1—LWR/LWR (MOX–U-Pu)	Option 2—LWR/HWR (DUPIC)		
Reactor Power Production (100 GWe)								
LWR–UOX or HWR–UOX or HTGR–UOX	100 GWe LWR	100 GWe HWR	100 GWe HTGR	0	0	73 GWe LWR	60 GWe LWR	63 GWe LWR
LWR–MOX-U-Pu, or LWR-HWR	0	0	0	0	100 GWe LWR	27 GWe HWR	0	7 GWe LWR
Fast Advanced Recycling Reactor	0	0	0	0	0	0	40 GWe ARR	30 GWe ARR
LWR–ThOX/UOX	0	0	0	100 GWe LWR	0	0	0	0
Fuel Burnup at Discharge (GWd/MTHM)	51	21	100	149 (UOX) 75 (ThOX)	45	35 (UOX) 15 (HWR)	51 (LWR) 107 (ARR)	51 (LWR) 50 (LWR – MOX/Pu) 105 (ARR)

A.7.1 Thermal Spectrum Burnup Trends

The greatest amount of data on thermal spectrum burnup and its impacts is available for uranium oxide (UOX) fuels used in LWRs. Therefore, this section begins with the No Action Alternative of UOX-fueled LWRs, which coincides with the current U.S. commercial nuclear fleet. The impact of burnup on HWR, Thorium and MOX fuels would follow the same general trends.

Historical U.S. commercial reactor operations show a steady trend toward higher burnup (Finck 2007b). The average improvement over the last 20 years is about 1 GWd/MTIHM per year. If this trend continues, burnup levels by 2020 will approach 60 GWd/MTIHM.

Effects of Higher Burnup

- Higher burnup means longer periods between refueling outages. Reactor owners complete significant maintenance during the refueling outages—the longer the time between outages, the more maintenance that builds up. Currently these outages occur every 18 months. (Higher burnup could be achieved while maintaining current refueling schedules if the fuel was left in the reactor for more cycles.)
- High burnup fuels tend to exhibit a greater frequency of cladding failures, suggesting the upper limit for current cladding material and manufacturing processes is being reached for the reactor fuel power density being used today.
- Current enrichment facilities are only licensed to achieve a 5 percent fuel enrichment for commercial applications. While the average enrichment in today's fuels is below the 5 percent limit, enrichment between assemblies and within assemblies varies to support load balancing. Portions of fuel elements are now at the 5 percent limit. (New enrichment facilities could be designed and licensed for higher enrichment, or it may be possible to relicense older facilities for production at higher enrichment.)
- Higher burnup requires higher enrichment. The enrichment levels for 51 GWd/MTIHM and 100 GWd/MTIHM UOX fuel average 4.3 percent and 8.5 percent respectively. So while the higher burnup produces approximately double the energy per unit of fuel, the increase in enrichment also is approximately double.

Higher burnup results in less SNF per unit of energy produced. The amount of SNF produced varies inversely to the SNF burnup—using the same reactor with the same thermal efficiency, increasing burnup by 50 percent decreases SNF by 33 percent (1.5 to 1.0) and doubling the burnup cuts SNF in half (2.0 to 1.0). The relationship is roughly linear over this range. This equates to fewer shipments of fresh fuel, but SNF shipments may not decrease linearly (or at all) due to the higher radiation and decay heat levels in the SNF. If the SNF shipments are volume limited, then half as much SNF may mean half as many shipments. But, if they are limited by decay heat or shielding requirements (or by weight, which is often driven by shielding), then the number of SNF shipments may not be reduced significantly. This is because, with higher burnup, the same number of fissions has taken place, but within fewer fuel assemblies—and those fewer assemblies still contain the same total masses of highly radioactive fission products. Since the fission products are more concentrated, each assembly is hotter (both radioactively and thermally) approximately in proportion to the increase in burnup. Appendix E of the PEIS indicates SNF shipments are limited by both volume and thermal considerations. The thermal impact can be somewhat reduced by storing the fuel longer before shipping, giving more time for the shorter-term fission products to decay.

Higher burnup also results in higher levels of heavier elements in the SNF. This is because there is more time for multiple neutron captures by the uranium atoms. Some of the created isotopes subsequently fission, such as some of the highly fissionable Pu-239 that is created through neutron capture by fertile U-238. This contributes to improved uranium utilization (since both U-235 and Pu-239 [produced from U-238] contribute to power production by fissioning). However, other heavier isotopes resulting from the capture of additional neutrons are typically long-lived and therefore contribute to both long-term decay heat and long-term radiotoxicity.

These heavier isotopes and their initial decay products include the transuranics, additional uranium isotopes, and other actinides.

In summary, for UOX in the once-through case, for equivalent energy production higher burnup results in lower amounts of SNF (in terms of MTHM) in proportion to the increase in burnup, but roughly the same amount of natural uranium is required and each SNF fuel assembly is both hotter and more radiotoxic, not quite in proportion to the increase in burnup. As a result, there is limited benefit for direct disposal of the SNF with increasing burnup. Transportation shipments may not be reduced due to thermal limits on SNF shipments and the potential for greater shipping requirements on the fuel cycle front end (since very high burnups require more natural uranium per unit of energy produced, more ore and UF_6 shipments would be needed). If the UOX fuel is recycled, the mass of radioactive materials in HLW is roughly the same per unit of energy produced (but much higher per MT of SNF). The amount of GTCC waste attributable to the cladding and assembly hardware scales closely with the MT of SNF recycled.

A.7.2 Fast Spectrum Burnup Trends

While the increase in burnup for the LWR was achieved by only increasing the initial fuel enrichment, the increase in burnup for the fast spectrum data was achieved either: 1) by increasing the mass of the core, reducing power density, and keeping about the same enrichment; or 2) by varying the enrichment, conversion ratio, and power density. Either approach makes it difficult to obtain results for all of the parameters of interest. The apparent trends related to fast spectrum burnup are as follows:

- The amount of uranium needed per equivalent energy production was not analyzed because for a burner fast reactor it is dependent on the LWR SNF that provides the transuranics and whether the uranium in the fast reactor fuel is recycled, natural, or depleted.
- SNF production per equivalent energy production trended similar to the thermal spectrum as burnup increased, but the changing heavy metal mass in the core makes the comparison difficult.
- Long term heat per equivalent energy production shows the same slightly downward trend as the other spectrums when varying the fluence[17] limit, but was essentially unchanged when varying the conversion ratio. In both cases the calculation is based on the material that would be disposed after reprocessing, assuming 0.1 percent loss of actinides.
- Long term radiotoxicity was not assessed but should trend similar to long term heat.

The general conclusion on the effect of discharge burnup on spent fast reactor fuel is that the HLW resulting from processing will be relatively unaffected by changes in burnup, with the result that disposal needs will be unaltered, although handling, storage, and shipping may be affected in the same manner as for the other cases.

[17] Fluence refers to the number of radiation particles crossing a given area.

A.8 EXISTING REACTOR REPLACEMENTS

Commercial LWRs are licensed by the NRC for 40 years of operation. NRC regulations also allow owner/operators to apply for 20-year license extensions. Table A.8-1 presents the operating license date for the commercial LWRs in the United States. As shown in that table, some of the existing LWRs in the United States have applied for, and received, license extensions. Additionally, other LWRs are likely to apply for, and receive, license extensions. This PEIS assumes that all existing LWRs, regardless of whether or not they receive license extensions, would require replacement during the time period analyzed (through approximately 2060).

TABLE A.8-1—*Operating Licenses of United States Light Water Reactors*

Reactor Name	Reactor Unit	Operating License Expiration	Operation License Start Date	Years of Operation Expected[a]
Arkansas Nuclear	1	5/20/2034	5/21/1974	60
	2	7/17/2038	9/1/1978	60
Beaver Valley	1	1/29/2016	7/2/1976	40
	2	5/27/2027	8/14/1987	40
Braidwood	1	10/17/2026	7/2/1987	40
	2	12/18/2027	5/20/1988	40
Browns Ferry	1	12/20/2033	12/20/1973	60
	2	6/28/2034	8/2/1974	60
	3	7/2/2036	8/18/1976	60
Brunswick	1	9/8/2036	11/12/1976	60
	2	12/27/2034	12/27/1974	60
Byron	1	10/31/2024	2/14/1985	40
	2	11/6/2026	1/30/1987	40
Callaway	1	10/18/2024	10/18/1984	40
Calvert Cliffs	1	7/31/2034	7/31/1974	60
	2	8/13/2036	8/13/1976	60
Catawba	1	12/5/2043	1/17/1985	59
	2	12/5/2043	5/15/1986	58
Clinton	1	9/29/2026	4/17/1987	40
Columbia	1	12/20/2023	4/13/1984	40
Comanche Peak	1	2/8/2030	4/17/1990	40
	2	2/2/2033	4/6/1993	40
Cooper Station	1	1/18/2014	1/18/1974	40
Crystal River	3	12/3/2016	1/28/1977	40
Davis-Besse	1	4/22/2017	4/22/1977	40
Diablo Canyon	1	11/2/2024	11/2/1984	40
	2	8/20/2025	8/26/1985	40
Donald C. Cook	1	10/25/2034	10/25/1974	60
	2	12/23/2037	12/23/1977	60
Dresden	2	12/22/2029	2/20/1971	60
	3	1/12/2031	1/12/1971	60
Duane Arnold	1	2/21/2014	2/22/1974	40
Enrico Fermi	2	3/20/2025	7/15/1985	40
Joseph Farley	1	6/25/2017	6/25/1977	40
	2	3/31/2021	3/31/1981	40
Fitzpatrick	1	10/17/2014	10/10/1974	40
Fort Calhoun	1	8/9/2033	8/9/1973	60
Grand Gulf	1	11/1/2024	10/1/1984	40

TABLE A.8-1—*Operating Licenses of United States Light Water Reactors (continued)*

Reactor Name	Reactor Unit	Operating License Expiration	Operation License Start Date	Years of Operation Expected[a]
H. B. Robinson	1	7/31/2030	9/23/70	60
Edwin Hatch	1	8/6/2034	10/13/1974	60
	2	6/13/2038	6/13/1978	60
Hope Creek	1	4/11/2026	7/25/1986	40
Indian Point	2	9/28/2013	9/28/1973	40
	3	12/12/2015	12/12/1975	40
Kewaunee	1	12/21/2013	12/21/1973	40
LaSalle County	1	4/17/2022	4/17/1982	40
	2	12/16/2023	12/16/1983	40
Limerick	1	10/26/2024	8/8/1985	40
	2	6/22/2029	8/25/1989	40
McGuire	1	6/12/2041	7/8/1981	60
	2	3/3/2043	5/27/1983	60
Millstone	2	7/3/2035	9/26/1975	60
	3	11/25/2045	1/31/1986	60
Monticello	1	9/8/2030	1/9/1971	60
Nine Mile Point	1	8/22/2029	12/26/1974	60
	2	10/31/2046	7/2/1987	60
North Anna	1	4/1/2038	4/1/1978	60
	2	8/21/2040	8/21/1980	60
Oconee	1	2/6/2033	2/6/1973	60
	2	10/6/2033	10/6/1973	60
	3	7/19/2034	7/19/1974	60
Oyster Creek	1	4/9/2009	7/2/1969	40
Palisades	1	3/24/2031	2/21/1971	60
Palo Verde	1	6/1/2025	6/1/1985	40
	2	4/24/2026	4/24/86	40
	3	11/25/2027	11/25/1987	40
Peach Bottom	2	8/8/2033	10/25/1973	60
	3	7/2/2034	7/2/1974	60
Perry	1	3/18/2026	11/13/1986	40
Pilgrim	1	6/8/2012	9/15/1972	40
Point Beach	1	10/5/2030	10/5/1970	60
	2	3/8/2033	3/8/1973	60
Prairie Island	1	8/9/2013	4/5/1974	40
	2	10/29/2014	10/29/1974	40
Quad Cities	1	12/14/2032	12/14/1972	60
	2	12/14/2032	12/14/1972	60
River Bend	1	8/29/2025	11/20/1985	40
Robert E Ginna	1	9/18/2029	9/19/1969	60
Salem	1	8/13/2016	8/13/1976	40
	2	4/18/2020	5/20/1981	40
San Onofre	1	2/16/2022	2/16/1982	40
	2	11/15/2022	11/15/1982	40
Seabrook	1	3/15/2030	3/15/1990	40
St. Lucie	1	3/1/2036	3/1/1976	60
	2	4/6/2043	6/10/1983	60
Sequoyah	1	9/17/2020	9/17/1980	40
	2	9/15/2021	9/15/1981	40
Shearon Harris	1	10/24/2026	1/12/1987	40
South Texas Project	1	8/20/2027	3/22/1988	40
	2	12/15/2028	3/28/1989	40

TABLE A.8-1—*Operating Licenses of United States Light Water Reactors (continued)*

Reactor Name	Reactor Unit	Operating License Expiration	Operation License Start Date	Years of Operation Expected[a]
Virgil C. Summer	1	8/6/2042	11/12/1982	60
Surry	1	5/25/2032	4/6/1973	60
	2	1/29/2033	1/29/1973	60
Susquehanna	1	7/17/2022	11/12/1982	40
	2	3/23/2024	6/27/1984	40
Three Mile Island	1	4/19/2014	4/19/1974	40
Turkey Point	3	7/19/2032	7/19/1972	60
	4	4/10/2033	4/10/1973	60
Vermont Yankee	1	3/21/2012	2/28/1973	39
Vogtle	1	1/16/2027	3/16/1987	40
	2	2/9/2029	3/31/1989	40
Waterford	3	12/18/2024	3/16/1985	40
Watts Bar	1	11/9/2035	2/7/1996	40
Wolf Creek	1	3/11/2025	6/4/1985	40

Source: NRC 2007i

[a]Assumes operation from start of license to expiration and no more than one license extension; does not presume that reactors which have not applied for, or have been granted a license extension, will operate past current license expiration date.

A.9 ADVANCED FUEL CYCLE INITIATIVE—SITES AND FACILITIES

As discussed in Chapter 2, the objective of the AFCI is to develop the technologies needed to: reduce the environmental consequences associated with spent nuclear fuel management, reduce the proliferation risk from the use of nuclear power, and extend uranium resources. Key elements of the initiative include:

- An Integration task, focused on providing overall consistency for the program and directing modeling and simulation and regulatory efforts for all tasks.
- A Systems Analysis task focused on investigating the interactions between program elements, evaluating deployment scenarios for various technical options, and identifying criteria that technologies would meet to allow the overall system to function effectively.
- A Separations task to develop and demonstrate advanced separations technologies for processing SNF, with an emphasis on LWR SNF.
- A Fuels task to develop and demonstrate transmutation fuels (including clad material) that could be used to destroy transuranic elements.
- A Waste Forms task to verify the long-term behavior of existing waste forms and develop new waste forms.
- A Safeguards task to develop and demonstrate new detection technologies and integrate them into high sensitivity nuclear protection systems.
- A Grid Appropriate Reactor task to develop and demonstrate small reactors that could be used in foreign countries with limited infrastructures. Such reactors also might be used within the United States. This task is an analytical activity that would eventually require the use of experimental facilities.
- A Reactor task to develop and demonstrate sodium-cooled fast reactor technologies that could be used for transmutation of nuclear wastes.

The AFCI relies on existing facilities, located mostly within DOE national laboratories.

The facilities described in the following sections are multi-purpose facilities, so only a portion of the work performed in the facilities is associated with the AFCI. For the purposes of estimating the environmental impacts of the AFCI Program in this PEIS, it is assumed that 15 percent of the activities that take place in the facilities, and therefore 15 percent of the worker dose and 15 percent of the waste generated by the facilities, are associated with the AFCI. This assumption was made based on an estimate of the percentage of known AFCI activities and their associated impacts ongoing at the Materials and Fuels Complex (MFC) facility at INL.

A.9.1 Argonne National Laboratory

ANL has primary responsibility for the AFCI fast reactor development, waste form development, and modeling and simulation research. This research is supported by the laboratory's Sodium Test Loop Facility. Some additional activities associated with AFCI are performed in other ANL facilities used by the laboratory's Nuclear Engineering Division, but the Sodium Test Loop work is directly related to AFCI research.

A.9.1.1 *Sodium Test Loop*

Basic description of facility and type of work performed:

The purpose of the sodium test loop is to determine the plugging effect of flowing sodium in small channels similar to the small channels in advanced heat exchangers, such as printed circuit heat exchangers. Radiological materials are not used in the sodium test loop.

The Sodium Test Loop Facility consists of a main sodium loop including three test sections, a bypass sodium loop including a cold trap/economizer assembly, and an auxiliary system comprising argon and vacuum lines. The main loop and the bypass loop are constructed from ½-inch (1.2 cm) thick, type 316 stainless steel tubing. Other major components include three electromagnetic flow controllers (one each for the three test sections), two electromagnetic pumps (one for the main loop and one for the bypass loop), five electromagnetic flow meters, and expansion and dump tanks.

The entire apparatus except the auxiliary system is placed over a stainless steel drip pan (100 in x 56 in x 2 in [2540 mm x 1422 mm x 51 mm]). The sodium loop system is about 5.9 ft (1.8 m) tall and is heated by a number of ceramic band heaters. The capacity of the system is approximately 1–2 gallons of sodium.

Number of employees in facility:

The sodium test loop is a small experimental loop located in a large highbay structure in Building 370 at ANL. The sodium test loop has approximately three to four personnel that support the work, two experimentalists and two technicians (all part time).

Types and quantities of radioactive wastes generated:

There are no radioactive wastes generated as a result of work conducted in the sodium test loop.

Significant radionuclide emissions:

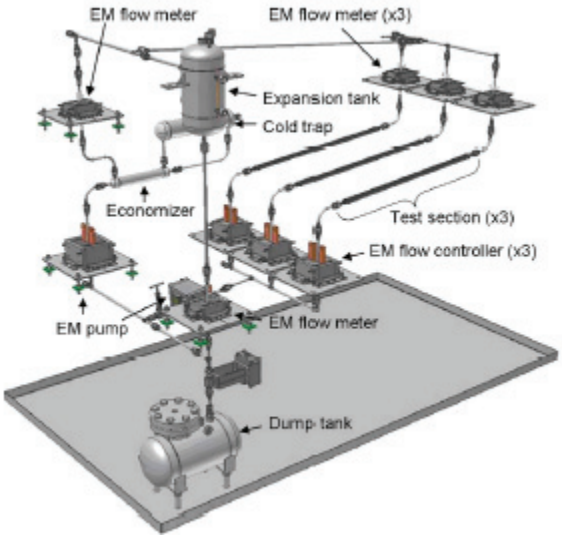
There are zero radionuclide emissions because radiological materials are not used in the sodium test loop.

Average person-rem for worker in facility:

None

Information on age of facility and plan for future use:

The test loop was placed into operation during 2008 and will be used to perform AFCI experimental work for the foreseeable future.



Source: DOE 2008d

FIGURE A.9.1.1-1—*Sodium Test Loop Schematic*

A.9.2 Hanford Site

Pacific Northwest National Laboratory (PNNL) performs a wide range of AFCI research, including significant activities associated with advanced fuel development and waste form development, at DOE's Hanford Site (Hanford). Most of this AFCI research is performed at the site's 300 Area and the Applied Process Engineering Laboratory (APEL).

A.9.2.1 *300 Area*

Basic description of facility and type of work performed:

The 300 Area is comprised of approximately 700,000 ft^2 (65,000 m^2) of space including approximately 50 percent of the PNNL's experimental space and all of its nuclear and radiological facilities. Four research and ancillary support buildings (Buildings 318, 325, 331, and 350) would be retained in the 300 Area if the PNNL Capability Replacement Laboratory

project is implemented as planned. The Capability Replacement Laboratory project involves potential construction of nearly 350,000 square feet of modern laboratories by 2011 and extending the operating life of Building 325 in the 300 Area by 20 years.

Buildings 318, 325, 331, and 350 are in relatively good condition but would require some upgrades. Building 325 is the Radiochemical Processing Laboratory where PNNL performs analytical radiochemistry verification and research support for complex materials. Building 331 is the Life Sciences Laboratory where PNNL performs a variety of biological sciences experiments. Building 318 is the Radiological Calibrations Building where PNNL performs calibration of radiation detection instruments and develops radiation dosimeters. Building 350 is the Plant Operations and Maintenance Facility that houses various shops.

As part of the Capability Replacement Laboratory project at PNNL, the Radiochemical Processing Laboratory is undergoing a series of upgrades to support future DOE missions. The upgrades include refurbishments of existing hot cells (including clean out and window replacement), six glovebox additions, and five modular hot cell additions with controlled atmosphere and inert gas capability to support irradiated material examinations. These upgrades also include facility infrastructure upgrades driven by the documented safety analysis for the facility, including ventilation blower modifications, seismic analysis and structural support, backup air compressor replacement, and contaminated exhaust duct replacement for worker dose control.

Of these facilities, the Radiochemical Processing Laboratory is best equipped to support AFCI missions of irradiated material examinations, separations research, and waste form development and qualification. The Radiochemical Processing Laboratory currently supports multiple program missions, including those of the Office of Science, National Nuclear Security Administration, Office of Environmental Management, and Office of Nuclear Energy. Currently less than 3 percent of the work performed at the facility is related to the AFCI. However, the facility has the capacity to support additional AFCI work. As a multi-program laboratory, future facility utilization is established and committed by contracts and work authorizations which are not approved until facility space and labor is reserved to perform the work.

Number of employees in facility:

Approximately 1,000 employees are directly linked to research currently performed in the 300 Area.

Types and quantities of radioactive wastes generated:

PNNL generates radioactive waste from its laboratory R&D and operational activities. The types of waste generated include LLW, and transuranic and mixed transuranic wastes. The wastes are typical of laboratory research and analytical processes. Radioactive wastes generated at PNNL are generally processed and/or packaged in waste management areas located within the Building 325 prior to shipment for final disposal. Listed below is an annual estimate of radioactive waste shipped for disposal (estimate is based on fiscal years 2005–2007 data). All wastes are managed in accordance with all applicable requirements.

LLW – 4,918 cubic feet (ft^3) (139 cubic meters [m^3])
Mixed LLW – 536 ft^3 (15 m^3)
TRU / Mixed TRU – 448 ft^3 (13 m^3)

Significant radionuclide emissions:

The PNNL contribution to dose associated with Hanford's radioactive air emissions was about 6.6×10^{-2} millirem (mrem) for Calendar year 2006 and represents greater than 99 percent of the point source emissions from Hanford. The primary isotope of concern is tritium from operations conducted in Building 325; tritium essentially makes up all of the offsite dose contribution. These releases are very small and within current permit limits, and none of the releases are associated with AFCI research, so it is estimated that no significant emissions have been produced by AFCI activities within the Hanford 300 Area.

Average person-rem for worker in facility:

The average dose to PNNL personnel, who received a measurable dose in 2006, was 73 mrem/year. The average dose to PNNL personnel, who were monitored in 2006, was 8.5 mrem/year. These should be considered approximate figures as actual doses vary from year to year, depending on activities.

Source: Burns 2008

FIGURE A.9.2.1-1—*Hanford Site 300 Area (Looking South to North)*

Source: Burns 2008

FIGURE A.9.2.1-2—*Radiochemical Processing Laboratory*

Information on age of facility and plan for future use:

The 300 Area is comprised of aging facilities, mostly built in the 1950s. PNNL is proposing that several new facilities be constructed on the main PNNL campus to replace 300 Area facilities.

PNNL is also proposing to build several new facilities within the 300 Area that could be used to support AFCI research. These facilities include:

- A Physical Sciences Facility—A 200,000 ft^2 (19,000 m^2) laboratory that would house radiological, materials science, and chemical research capabilities.
- A Computational Sciences Facility—A 75,600 ft^2 (7,000 m^2) facility that would house information analytics capabilities, computer laboratories, and electronics and instrumentation laboratories.

Construction and operation of the Physical Sciences Facility is analyzed in the *Environmental Assessment for the Construction and Operation of a Physical Sciences Facility at the Pacific Northwest National Laboratory,* (DOE/EA-1562) (DOE 2007e) and *Finding of No Significant Impact for the Construction and Operation of a Physical Sciences Facility at the Pacific Northwest National Laboratory* (DOE 2007ii).

A.9.2.2 Applied Process Engineering Laboratory

Basic description of facility and type of work performed:

The Applied Process Engineering Laboratory is an eastern Washington technology business startup user facility, sponsored in part by PNNL. APEL provides engineering- and manufacturing-scale space and chemical, biological, and electronic laboratories and equipment for developing, validating, and commercializing new products. Entrepreneurs, engineers, scientists, and university staff can access this facility. PNNL scientists, engineers, and other

professional staff are available to APEL occupants for consulting, collaboration, or professional support.

The facility has a single story office wing on the south section, two floors of laboratories in the center section (which are serviced by an elevator), and a 28-ft high bay along the north end with roll up door access.

The APEL facility houses the Glass Development Laboratory which is utilized by resident PNNL staff to design and test glass and ceramic waste forms. This laboratory is utilized in the AFCI project to develop and test waste forms for fission products that emerge from proposed spent fuel separations processes. Nonradioactive surrogates are used with sophisticated instrumentation and equipment designed to formulate, examine, and test various glass and ceramic waste forms.

Number of employees in facility:

The number of employees involved in AFCI research at the facility varies with the experiment being performed. Most experiments require fewer than 20 support personnel.

Types and quantities of radioactive wastes generated:

No radioactive wastes are generated as a result of AFCI research at the facility.

Significant radionuclide emissions:

There are no radionuclide emissions from the facility as a result of AFCI research.

Average person-rem for worker in facility:

AFCI research at the facility does not produce any worker dose.

Information on age of facility and plan for future use:

The APEL facility was built in 1975 and an existing structure was modified beginning in the summer of 1997 to contain wet labs, business startup bays, permitted high bay development space, and a computer center. The facility is expected to continue supporting AFCI research for the foreseeable future.

A.9.3 Idaho National Laboratory

INL has primary responsibility for AFCI research associated with systems analysis, advanced fuels development, and separations process development. INL's AFCI experimental research is performed at the Materials and Fuels Complex and Reactor Technology Complex.

A.9.3.1 *Materials and Fuels Complex*

Basic description of facility and type of work performed:

The Materials and Fuels Complex (MFC) is located on a 1,700-acre (690 ha) tract of the INL about 35 miles (mi) (56 kilometers [km]) west of Idaho Falls. MFC contains 595,150 ft^2 (55,300 m^2) of floor space. This campus is dedicated to performing R&D for reactor fuels, fuel cycle, and related materials.

Work at MFC predominantly involves fundamental research and technology development, with results disseminated openly and shared with the scientific community or made available to private industry. Other work includes training programs as well as Department of Homeland Security R&D. Other research involves commercial interests.

Facilities housing major capabilities have been maintained and renovated over the years to support the pursuit of nuclear energy initiatives such as the AFCI. However, the MFC continues to face significant challenges from the normal aging of buildings and infrastructure, and a substantial need for upgraded laboratory facilities. Various proposals for upgrading MFC facilities are under development or under consideration, but AFCI research will continue at MFC for the foreseeable future even if these plans are not implemented.

Number of employees in facility:

The facility currently employs about 700 people.

Types and quantities of radioactive wastes generated:

Wastes generated at MFC generally include personnel protective equipment, laboratory wastes, and other materials contaminated with low levels of radionuclides. Small volumes of waste are periodically generated that have higher radionuclide and chemical contaminant concentrations but these wastes are typically still classified as low-level waste or mixed waste. Demolition and construction activities at the site also periodically produce significant volumes of building and construction debris. All wastes from the facility are managed in accordance with all applicable requirements.

Significant radionuclide emissions:

All emissions from MFC are well within allowable limits. The effective dose equivalent to the maximally exposed individual member of the public from all INL sources was 9.3×10^{-2} millirem during 2007 (the latest year for which data is available) (DOE 2008i). This dose equivalent is less than the 10-millirem per year federal standard established under the National Emission Standards for Hazardous Air Pollutants (NESHAP).

Average person-rem for worker in facility:

Personnel exposure at MFC is maintained as low as reasonably achievable, and controls such as additional training and management reviews of exposure are typically implemented for worker exposures greater than 100 mrem/year. If 100 mrem/year is taken as the upper bound for average annual worker dose, and AFCI activities are assumed to account for 15 percent of worker dose at the facility, the upper bound average annual dose for AFCI work at MFC is estimated to be 15 mrem/year.

Source: Burns 2008

FIGURE A.9.3.1-1—*Materials and Fuels Complex at INL (Looking West to East)*

Information on age of facility and plan for future use:

Approximately 78 percent of MFC space is more than 30 years old. Planning for strategic modernization of MFC will focus on upgrading site-wide utilities, maintaining and modernizing major nuclear and radiological facilities, eliminating non-usable space and constructing modern facilities. Upgrade projects could include:

- Fuel fabrication facility upgrades
- Hot Fuel Examination Facility (HFEF) facility and infrastructure upgrades
- Transient Reactor Test Facility (TREAT) transient test restart and upgrade
- Fuel Cycle Facility (FCF) stack monitor and infrastructure upgrade

- FCF remote fuel fabrication upgrade
- Analytical laboratory upgrade

The capabilities at the MFC provide support to the AFCI program, from fuel fabrication and post irradiation examination to recycle technology development and demonstration. For this reason, the FCF, HFEF, FMF, TREAT, and analytical laboratory facilities are the facilities where the R&D activities of the AFCI program would be conducted.

A.9.3.2 *Reactor Technology Complex*

Basic description of facility and type of work performed:

The Reactor Technology Complex (RTC) is located on a 104-acre (42 ha) fenced complex on the INL Site about 47 mi (76 km) west of Idaho Falls, ID. RTC buildings contain approximately 620,000 ft^2 (57,600 m^2) of total floor space. The primary focus of RTC is continued operation of the Advanced Test Reactor (ATR) to conduct fuel and materials irradiation testing, nuclear safety research, and nuclear isotope production. These activities support development of advanced, safer, more efficient, and proliferation-resistant nuclear reactors, in addition to production of medical, commercial, and government-owned isotopes.

In April 2007, the DOE designated the ATR a National Scientific User Facility. As a Scientific User Facility, the ATR offers capabilities for nuclear fuel and reactor materials system development that universities, industry and regulatory agencies will be able to utilize. The ATR's core design allows many experiments to be conducted simultaneously, with each experiment receiving a different and carefully controlled level of radiation. Among the programs this research will support are AFCI and the Next Generation Nuclear Plant.

Much of the future work at RTC would involve research and technology development in collaboration with commercial and academic sectors, especially given the thrust to make ATR more accessible as a user facility. The environment necessary to efficiently foster such collaboration requires adapting the RTC area to a more campus-like layout, with enhanced access to commercial, academic, and foreign visitors and assignees, with whom active research partnering is encouraged.

Number of employees in facility:

The facility currently employs about 680 people.

Types and quantities of radioactive wastes generated:

Hazardous, LLW, and mixed LLW are generated at RTC and generally consist of personnel protective equipment, laboratory wastes, and other materials contaminated with low levels of hazardous materials and radionuclides. Small volumes of waste are periodically generated that have higher radionuclide and chemical contaminant concentrations. Demolition and construction activities at the site periodically produce significant volumes of building and construction debris. All wastes from the facility are managed in accordance with all applicable requirements.

Significant radionuclide emissions:

All emissions from RTC are well within allowable limits. The effective dose equivalent to the maximally exposed individual member of the public from all INL sources was 9.3 x 10^{-2} millirem during 2007 (the latest year for which data is available) (DOE 2008i). This dose equivalent is less than the 10-millirem per year federal standard, established under the National Emission Standards for Hazardous Air Pollutants (NESHAP).

Average person-rem for worker in facility:

The average dose to 502 people who received greater than 1 mrem at RTC during 2004, 2005, and 2006 was 86 mrem. If it is assumed that 15 percent of worker dose at RTC is associated with AFCI activities, an upper bound estimate of the average annual worker dose associated with AFCI activities at RTC is approximately 13 mrem/year.

Source: Burns 2008

FIGURE A.9.3.2-1—*Reactor Technology Complex at INL (Looking South to North)*

Information on age of facility and plan for future use:

The majority of RTC facilities are more than 40 years old, with a high degree of deferred maintenance. While the ATR Life Extension Project, and other efforts, are beginning to address the need for enhanced investment in the primary ATR systems and capabilities, a substantial need remains for investment in maintenance and upgrading of major ATR support systems, facilities, and utilities. However, this situation also presents an opportunity to provide modern capabilities while eliminating dilapidated space and reducing maintenance liabilities through

footprint reduction, space optimization, and strategic infrastructure investments. Primary upgrade needs include:

- Fast Spectrum Gas Test Loop upgrade
- ATR Life Extension Project

These upgrades would directly support the AFCI fuel irradiation and testing program. It is expected that over the duration of the AFCI the gas test loop would be used extensively by the program. Following the completion of the AFCI irradiation program however, the gas test loop would be available to other programs for use. The life extension project is a general support project that benefits all users of the ATR and as such approximately 20 percent of the utilization of the ATR could be attributed to the AFCI irradiation program over the duration of the project. Additional impacts of utilization at 20 percent versus the 15 percent currently utilized would be minimal.

A.9.4 Los Alamos National Laboratory

Los Alamos National Laboratory (LANL) supports all areas of AFCI research including major activities associated with advanced fuel development and safeguard system research. Facilities used to perform this research include the Chemistry and Metallurgy Research Building, the Los Alamos Neutron Science Center, Technical Area–35 (TA-35), and Technical Area–55 (TA-55).

A.9.4.1 *Chemistry and Metallurgy Research Building*

Basic description of facility and type of work performed:

The Chemistry and Metallurgy Research (CMR) Building is a research and experimental facility for analytical actinide chemistry, metallurgy, and materials science. The facility, located in Technical Area-3 (TA-3), currently houses research and experimental laboratories for analytical chemistry and plutonium and uranium chemistry and metallurgy.

CMR covers approximately 550,000 ft^2 (51,100 m^2) divided among three stories. CMR consists of seven laboratory wings including one with heavily shielded hot cells with remote handling capabilities. CMR is the only LANL facility with full capability for analytical chemistry and metallurgical studies on small samples of plutonium and other special nuclear materials. Analyses performed at CMR are critical to DOE defense programs since they help ensure plutonium pit production and testing specifications are met.

Number of employees in facility:

More than 280 people are employed in the CMR area.

Types and quantities of radioactive wastes generated:

The majority of research activities within the CMR laboratories involve various isotopes of uranium and plutonium. Other radionuclides, including mixed fission products, have also been used in the laboratories.

LLW arising from standard operations at CMR consists primarily of contaminated and potentially contaminated laboratory equipment, personnel protective equipment, and laboratory waste products. Construction upgrades within the building also produce LLW in the form of construction and building debris. All wastes from the facility are managed in accordance with all applicable requirements.

Radiological contamination:

Leaks and spills over the history of the CMR have resulted in various degrees of fixed and removable contamination. Alpha contamination found in the facility is typically the result of past spills of plutonium contaminated materials and beta/gamma contamination found in the facility is typically associated with past spills of uranium contaminated materials. These areas are decontaminated to the maximum extent possible when they are identified, but activities in CMR must be managed understanding the potential for radioactive contamination.

Wastes generated at CMR consist of hazardous waste, LLW, mixed LLW, TRU waste, and non-hazardous solid waste. These wastes are disposed of in accordance with all applicable requirements. The majority of the LLW is managed on-site with some quantities being shipped off-site to NTS and commercial LLW disposal facilities. The hazardous waste is shipped off-site to RCRA licensed commercial waste management facilities.

Emissions of radioactive particulate matter are sampled using a glass-fiber filter. A continuous sample of stack air is pulled through the filter, where small particles of radioactive material are captured. These samples are analyzed weekly using gross alpha/beta counting and gamma spectroscopy to identify any increase in emissions and to identify short-lived radioactive materials. Every six months, LANL takes composites of these samples for analysis at an off-site Laboratory. These composite samples are analyzed to determine the total activity of materials such as uranium-234/235/238, plutonium-238/239/240, and americium-241. These data are then combined with estimates of sampling losses and stack and sample flows to calculate emissions (LA-13732-ENV).

Average person-rem for worker in facility:

The average annual non-zero worker dose produced by activities within CMR is less than 100 mrem/year. If 100 mrem/year is taken as the upper bound for average annual worker dose, and AFCI activities are assumed to account for 15 percent of worker dose at the facility, the upper bound average dose for AFCI work at CMR is estimated to be 15 mrem/year.

Picture/schematic of facility:

Source: Burns 2008

FIGURE A.9.4.1-1—*Operation of Hot Cell Equipment Within the Chemistry and Metallurgy Research Building*

Information on age of facility and plan for future use:

The CMR, located within TA-3, at LANL, consists of seven wings that were constructed in 1952; a new wing was added in 1960 for activities that must be performed in hot cells. The three story building is a multiple-user facility in which specific wings are associated with different activities. It is the only LANL facility with full capabilities for performing special nuclear material analytical chemistry and materials science.

The National Nuclear Security Administration (NNSA) issued a Record of Decision (ROD) on the proposed replacement of the existing CMR Building in 2004. The ROD was based upon the information contained in the *Environmental Impact Statement for the Proposed Chemistry and Metallurgy Research Building Replacement Project, Los Alamos National Laboratory, Los Alamos, New Mexico*, DOE/EIS-0350 (CMRR EIS), and documented the decision to build a CMR Replacement (CMRR) facility at TA-55 (DOE 2003c). The new CMRR facility would include a single, above-ground, consolidated special nuclear material-capable laboratory building (known as the CMRR-Nuclear Facility), with a separate administrative office and support functions building. The existing CMR building at LANL would be decontaminated, decommissioned, and demolished in its entirety.

NNSA is continuing the preliminary design of the CMRR nuclear facility. NNSA will decide whether to construct this nuclear facility after completion of the *Complex Transformation Supplemental Programmatic Environmental Impact Statement (SPEIS)* (DOE/EIS-0236-S4) and subsequent decisions regarding plutonium manufacturing and R&D activities (DOE 2007z). Should a site other than LANL be selected, the CMRR might still be constructed at LANL in order to provide metallurgy chemical activities in support of an interim pit production capability until a new pit production facility is available, as well as to support the special metallurgical needs of other DOE programs, such as AFCI. In any case, NNSA has determined that preliminary design of the CMRR nuclear facility would be applicable to any future pit production facility at any site analyzed in the Complex Transformation SPEIS.

A.9.4.2 *Los Alamos Neutron Science Center*

Basic description of facility and type of work performed:

The Los Alamos Neutron Science Center (LANSCE) lies entirely within TA-53. The facility has more than 400 buildings, including one of the largest at LANL (Building 53-3), which houses a linear accelerator and has 316,000 ft^2 (29,400 m^2) of floor space under roof. Activities at the facility consist of neutron science and nuclear physics research, proton radiography, the development of accelerators and diagnostic instruments, and production of medical radioisotopes.

The majority of the LANSCE facility is composed of the 800-million electron-volt linear accelerator, a Proton Storage Ring, and three major experimental areas: the Manuel Lujan Neutron Scattering Center, the Weapons Neutron Research (WNR) facility, and Experimental Area C. Experimental Area C is the location of proton radiography experiments for the Stockpile Stewardship Program. A new experimental facility for the production of ultracold neutrons is nearing completion in Area B. Experimental Area A, formerly used for materials irradiation experiments and isotope production, is currently inactive; construction of a new Isotope Production Facility was completed in CY 2002 and commissioning occurred in 2003.

The LANSCE facility includes three nuclear activities: experiments using neutron scattering by transuranics in Experimental Area ER-1/ER-2, the 1L neutron production target in Building 53-7, and Area A East in Building 53-3M, which is used for passive storage of activated materials. In 2001, TA-53-945 and 53-954 were placed on the LANL Radiological Facility List (LANL 2002). Facilities on this list satisfy the definition of radiological facility. Experimental Area ER-1/ER-2 is categorized as a Moderate Hazard facility. The remainder of the LANSCE facility is categorized as Low Hazard.

Number of employees in facility:

Tens to hundreds of people work in the LANSCE facility depending on scheduled work activities.

Types and quantities of radioactive wastes generated:

The LANSCE facility includes four National Pollutant Discharge Elimination System (NPDES) permitted outfalls associated with operation of the facility's cooling towers (three in Los Alamos Canyon and one in Sandia Canyon). The outfalls produce a total estimated annual discharge of 29.5 million gallons (111.7 million liters) per year (DOE 2008c). Radionuclide concentrations in the outfall water are typically less than detectable concentrations.

The LANSCE facility produces approximately 1,400 cubic yards (yd^3) (1070 cubic meters [m^3]) of LLW and 1 yd^3 (0.8 m^3) of mixed LLW per year (DOE 2008c). The waste is disposed of at permitted onsite or offsite disposal facilities. The facility also produces approximately 135,000 gallons (511,000 liters) of radioactive liquid waste per year that are treated at the facility. All wastes are managed in accordance with all applicable requirements.

Significant radionuclide emissions:

The emission stacks at the LANSCE facility in TA-53 are the primary source of LANL airborne radionuclide emissions. However, emissions from the stack meet current permit requirements. Estimates of annual emissions from the stacks are shown in Appendix C of the *Final Site-Wide Environmental Impact Statement for Continued Operation of Los Alamos National Laboratory, Los Alamos, New Mexico* (LANL Site-wide Environmental Impact Statement) (DOE 2008c).

Average person-rem for worker in facility:

The average annual non-zero worker dose produced by activities within the LANSCE facility is less than 100 mrem/year. If 100 mrem/year is taken as the upper bound for average annual worker dose, and AFCI activities are assumed to account for 15 percent of worker dose at the facility, the upper bound average dose for AFCI work at the LANSCE facility is estimated to be 15 mrem/year.

Picture/schematic of facility:

Source: Burns 2008

FIGURE A.9.4.2-1—*The Los Alamos Neutron Science Center (Looking East to West)*

Information on age of facility and plan for future use:

Since the LANSCE linear accelerator first accelerated protons in 1972, the facility mission has evolved considerably. However, investment in the physical infrastructure and technology has not been adequate to ensure long-term sustainable operation at high reliability. The LANSCE Refurbishment Project has received preliminary funding for design and is on schedule to progress to the architectural design stage during fiscal year 2008. The project proposes to sustain reliable facility operations well into the next decade. The LANSCE Refurbishment Project will address the following priorities: 1) replacing facility equipment where necessary to address code compliance or end-of life issues that could severely impact facility operations; 2) enhancing cost-effectiveness by system refurbishments or improvements that stabilize decreasing facility reliability and maintainability; 3) stabilizing the overall beam availability and reliability in a manner that is sustainable over the longer term; and 4) accomplishing the above with minimal disruption to scheduled user programs. The LANSCE Refurbishment Project could support the AFCI Program by providing proton beam power and reliability needed to support testing and qualification of advanced reactor fuels.

Upgrade work associated with the Materials Test Station (MTS) is also planned at LANSCE. The MTS is part of a fast neutron source capability project that has received preliminary approval from DOE. The project is expected to produce a non-reactor source of fast neutrons that can be used for long-term irradiation testing to support AFCI experiments.

A.9.4.3 *Technical Area - 35*

Basic description of facility and type of work performed:

Technical Area–35 (TA-35) is located in the north central portion of LANL. The facility is used for nuclear safeguards R&D, primarily in the areas of lasers, physics, and materials development. Additional activities at TA-35 include research in reactor safety, optical science, and pulsed-power systems, as well as metallurgy, ceramic technology, and chemical plating. The facility was formerly the site of the Atlas project; the Atlas removal project has been completed, and the building is now available as storage space. Additionally, there are some Biosafety Level 1 and 2 laboratories at TA-35.

The Target Fabrication Facility, located at TA-35, conducts precision machining and target fabrication, polymer synthesis, and chemical and physical vapor deposition. The facility is comprised of three buildings (35-213, 35-455, and 35-458). The main building is a two-story structure encompassing approximately 61,000 ft^2 (5,700 m^2) of floor space housing activities related to weapons production and laser fusion research. The Target Fabrication Facility is located immediately to the east of Technical Area–55 (TA-55) and directly north of Technical Area–50 (TA-50). This facility is categorized as a low hazard non-nuclear facility, although low levels of radioactive material are used. Exhaust air from process equipment is filtered prior to exhaust to the atmosphere. Sanitary waste is piped to the sanitary waste disposal plant located in Technical Area–46 (TA-46). Radioactive liquid waste and liquid chemical waste are transported to the TA-50 Radioactive Liquid Waste Treatment Facility using a direct pipeline.

The Nuclear Safeguards Research Building is a three-story sheet metal, steel, and concrete block building containing approximately 45,000 ft^2 (4,200 m^2) of floor space. Levels 2 and 3 are underground. The primary activities in Building 27 are nuclear safeguards research, development, and training, which address new ways of conducting nondestructive analysis tests on samples of many different sizes and shapes to determine their uranium and plutonium content. This R&D is supported by electronics development, mechanical design and fabrication, and administrative activities. All radioactive sources and special nuclear materials within the building are encapsulated to prevent any contamination of workers or the facility. The uranium in the facility is singly contained, and plutonium is doubly contained. No nuclear material is processed, and samples remain sealed at all times, including when they are used in instruments. Special nuclear material is used as a radiation source for calibrating and testing the performance of prototype and finished instruments, as well as for the Nuclear Safeguards Technology Training Program.

Number of employees in facility:

More than 165 people are employed at TA-35.

Types and quantities of radioactive wastes generated:

Approximately 13 yd^3 (10 m^3) of LLW and 0.5 yd^3 (0.4 m^3) of mixed LLW are generated at TA-35 per year (DOE 2008c).

Significant radionuclide emissions:

Contaminants have been released from outfalls, air stack emissions, and cooling water and septic system discharges at TA-35. From 1951 until 1963, the wastewater treatment facility discharged effluent into Ten Site Canyon. Spills occurred from leaks in pipelines, structures, and container storage areas. Potential contaminants include metals, PCBs, volatile organic compounds, and radionuclides (NMED 2006).

Wastes currently generated at TA-35 are managed accordance with all applicable requirements.

Average person-rem for worker in facility:

The average annual non-zero worker dose produced by activities within TA-35 is less than 100 mrem/year. If 100 mrem/year is taken as the upper bound for average annual worker dose, and AFCI activities are assumed to account for 15 percent of worker dose at the facility, the upper bound average dose for AFCI work at TA-35 is estimated to be 15 mrem/year.

Information on age of facility and plan for future use:

TA-35 has been in operation for nearly 30 years. Nuclear research activities are expected to continue at the site for the foreseeable future.

A.9.4.4 *Technical Area - 55*

Basic description of facility and type of work performed:

Facilities in TA-55 support pit manufacturing and surveillance and special plutonium recovery. To meet the varied needs of research, development, and plutonium processing programs at LANL, TA-55 provides chemical and metallurgical processes for recovering, purifying, and converting plutonium and other transuranics into many compounds and forms. Additional capabilities include the means to safely and securely ship, receive, handle, and store nuclear materials, as well as manage the waste and residues produced by TA-55 operations.

Core capabilities include basic and applied research into the chemistry of plutonium and other transuranics, study of nuclear materials, and a strong technology base in nuclear materials separations, processing, and recovery. The facility also supports research in plutonium metallurgy; actinide surface studies; plutonium-component fabrication technologies, including pit manufacturing and surveillance; and actinide ceramics for heat sources and MOX fuels.

Core competencies are maintained for every type of plutonium-processing activity along with the ability to convert recovered material into plutonium metal. In addition, analytical capabilities, techniques for materials control and accountability, and substantial R&D capabilities are available to support these core competencies.

TA-55 houses a sophisticated system for nuclear materials accounting, management, and modeling; a measurement support operation; and a non-destructive assay laboratory. A nuclear

materials packaging and transfer operation receives nuclear material at the facility and transfers shipments from the facility. Safe nuclear materials storage is also provided. All nuclear materials in process or stored on site are monitored to ensure that material balances are properly maintained and inventoried on a real-time basis. A small portion of these capabilities, mostly associated with ceramic fuel development work, are used to support the AFCI program.

Number of employees in facility:

More than 1,000 people are employed at TA-55.

Source: Burns 2008

FIGURE A.9.4.4-1—*Technical Area - 55 at LANL (Looking West to East)*

Types and quantities of radioactive wastes generated:

Virtually all items currently disposed as LLW from TA-55 have been contaminated with plutonium and plutonium progeny (primarily Am-241). The only exceptions have been for disposal of spent radioactive calibration sources. Most waste items are described as either compactable boxed room trash (e.g., small laboratory items, personnel protective equipment,

paper, etc.) or building debris (e.g., wood, plastic, metal, paper, rubber, glass, waste rags, absorbed liquids, equipment, concrete, and other general building debris).

Radioactive waste management:

Wastes generated at TA-55 are managed in accordance with all applicable requirements. A significant portion of waste management operations take place in facilities designed for and dedicated to waste management. Liquid wastes are treated in the Sanitary Wastewater Systems Plant, the High Explosives Wastewater Treatment Facility, and the Radioactive Liquid Waste Treatment Facility. Specialized facilities in Technical Area–50 (TA-50) and Technical Area–54 (TA-54) house a variety of chemical and radioactive waste management operations, including size reduction, compaction, assaying, and storage. Many hazardous wastes are now accumulated for up to 90 days at consolidated storage facilities and are then shipped directly offsite to RCRA permitted commercial waste facilities. Four of these consolidated storage facilities exist at LANL and two more are planned (DOE 2007z).

Most low-level radioactive waste generated at LANL is disposed onsite at TA-54, Area G. Disposal operations were expanded into Zone 4, providing sufficient capacity for operational wastes for the long term. Although there were several instances of individual facilities exceeding 1999 projections, overall LANL low-level radioactive waste generation was well below those levels predicted in 1999 for five years of the six-year period (DOE 2007z).

Average person-rem for worker in facility:

The average annual non-zero worker dose produced by activities within TA-55 is less than 100 mrem/year. If 100 mrem/year is taken as the upper bound for average annual worker dose, and AFCI activities are assumed to account for 15 percent of worker dose at the facility, the upper bound average dose for AFCI work at TA-55 is estimated to be 15 mrem/year.

Information on age of facility and plan for future use:

The TA-55 Plutonium Facility Complex consists of six primary buildings and a number of support, storage, security, and training structures located throughout the main complex at TA-55. Constructed in the mid-1970s, it has been in operation for approximately 30 years, and had a recent modification which became operational in November 2005. Although systems in this complex function as designed, many are near the end of their design lives and have become increasingly difficult and expensive to maintain. The National Nuclear Security Administration (NNSA) has determined that an investment is needed in the near term to upgrade electrical, mechanical, safety, and other selected facility-related systems that are approaching the end of life. Several construction projects and upgrades were planned for the Plutonium Facility Complex and analyzed in the 1999 LANL Site-wide Environmental Impact Statement. The upgrades considered in this document would support all programs conducted in the facility, including fabrication and characterization of advanced fuels for the AFCI.

A.9.5 Oak Ridge National Laboratory

The Oak Ridge National Laboratory (ORNL) is one of the three primary installations on the DOE Oak Ridge Reservation (ORR) in Oak Ridge, Tennessee. The other installations are the Y-12 National Security Complex and the East Tennessee Technology Park (formerly the K-25 Site). ORNL performs AFCI research associated with separations and advanced fuel development, and the laboratory has primary responsibility for AFCI grid appropriate reactor research. Oak Ridge facilities used to perform AFCI research include the Radiochemical Engineering Development Center (REDC) and the Irradiated Fuel Examination Laboratory (IFEL).

A.9.5.1 *Irradiated Fuel Examination Laboratory*

Basic description of facility and type of work performed:

Destructive testing of SNF is performed at the IFEL. The facility is used for receipt, segmentation, and testing of SNF, and includes equipment used for voloxidation experiments and fission gas capture. Fuel is typically processed through IFEL before being transferred to REDC for chemical processing.

Number of employees in facility:

Approximately 10 personnel work in IFEL on a full time basis.

Types and quantities of radioactive wastes generated:

Wastes generated at IFEL typically include:

- 0 to 10 ft^3 (0 to 0.3 m^3) per year of highly irradiated metal from fuel testing. The metal typically includes significant quantities of Co-60, iron isotopes, and aluminum isotopes.
- Approximately 400 to 500 ft^3 (11 to 14 m^3) per year of solid low level waste contaminated with mixed fission products.
- One 55-gallon (208-liter) drum of contact handled transuranic waste per year.
- Less than 2 ounces (50 grams) of SNF residue per year.

The majority of waste management facilities at Oak Ridge are operated by NNSA. Waste management facilities are located in buildings, or on sites, dedicated to their individual functions, or are colocated with other waste management facilities or operations. Many of the facilities are used for more than one waste stream. Hazardous waste is managed in accordance with all applicable regulatory requirements and is shipped off site to a RCRA permitted commercial facility. LLW and non-hazardous solid wastes are managed on site. TRU waste is packaged according to the Waste Isolation Pilot Plant (WIPP) Waste Acceptance Criteria and shipped to the WIPP for disposal.

Significant radionuclide emissions:

Air emissions, including 1 to 3 curies of radioactive gases per year, are typically discharged from the facility's stack. Air discharges comply with current permit levels.

Average person-rem for worker in facility:

Approximately 200 mrem/year

Information on age of facility and plan for future use:

The IFEL was built in 1964 and has been in continuous use since it began operation. The facility is expected to continue operation for the foreseeable future.

Source: Burns 2008

FIGURE A.9.5.1-1—*Exterior of the Irradiated Fuel Examination Laboratory*

A.9.5.2 *Radiochemical Engineering Development Center*

Basic description of facility and type of work performed:

The REDC at ORNL processes irradiated targets for research and industrial users. In a typical processing campaign, dekagram quantities of curium are recovered for recycle into targets for subsequent irradiation and processing, decigram quantities of californium are recovered for

fabrication into neutron sources, and milligram quantities of einsteinium and berkelium, as well as picogram quantities of fermium, are recovered for distribution to the research community.

The heart of the REDC is a battery of nine heavily shielded hot cells housed in a two-story building. Of the nine cells, four contain chemical processing equipment for dissolution, solvent extraction, ion exchange, and precipitation operations. Three contain equipment for the preparation and inspection of transuranic element targets. One cell is used for analytical chemistry operations, and another is used for waste collection and sorting.

The Coupled End to End (CETE) experiment is a demonstration that supports development of Uranium Extraction Plus (UREX+). The CETE experiment takes single rods from an irradiated SNF bundle, and performs a number of small-scale processes on the fuel. The processes performed in the demonstration include:

- Voloxidation
- Fuel Dissolution
- Off-gas capture and treatment
- Solvent extraction
- Product solidification (U or U/Pu or U/Pu/Np)

Waste products are collected, solidified (if necessary) and sent to other national laboratories including Savannah River National Laboratory, Sandia National Laboratory, Pacific Northwest National Laboratory, Los Alamos National Laboratory, and Idaho National Laboratory, for development and demonstration of immobilization processes and waste form characterization.

Number of employees in facility:

The facilities used for AFCI research are located in the complex with the High Flux Isotope Reactor (HFIR) and other isotope production facilities. The facility used for the head-end processing is also used for extensive fuels examination work. Several hundred employees work within the facilities but only about 60 to 70 percent of the employees are involved in AFCI research, on a full or part-time basis.

Types and quantities of radioactive wastes generated:

Depending on the flowsheet being used, the various streams are being collected for subsequent work on development of the waste forms. The project generates wastes all of which are managed in accordance with applicable requirements. The majority of the wastes generated at Oak Ridge are managed by NNSA. LLW is disposed of on-site, and hazardous materials are shipped off-site to RCRA licensed commercial waste management facilities.

Significant radionuclide emissions:

There are no significant radionuclide emissions from the facility. Iodine, krypton, xenon, and tritium are trapped within the facility for use in waste form development research.

Average person-rem for worker in facility:

Approximately 50 mrem/year

A.9.6 Sandia National Laboratories/New Mexico

Sandia National Laboratories/New Mexico (SNL/NM) has primary responsibility for AFCI activities associated with regulatory and safety research, and the laboratory supports fast reactor development. Research activities are performed at the laboratory's Brayton Cycle Demonstration Facility.

A.9.6.1 *Brayton Cycle Demonstration*

Basic description of facility and type of work performed:

Brayton cycles are promising power conversion systems for the higher outlet temperatures of advanced reactors but there is little experience with closed Brayton-cycle systems coupled to reactor systems. The Sandia Brayton Cycle Demonstration Facility was developed to provide a closed Brayton cycle test facility to investigate the mechanisms for control and system behavior under dynamically varying loads, during startup and shut down conditions, including the capabilities for near autonomous operation. The test-loop was first operational in fiscal year 2005 and is used to provide model validation data and simulate nuclear reactor operation with gas turbine power conversion systems. The current closed Brayton cycle test bed operates with a range of working fluids and mixtures at a nominal few bar pressures.

The turbo machinery is based on a commercially available Capstone micro-turbine power plant (30 kilowatt electric [kWe]). The Capstone open cycle gas turbine system was selected largely because it was readily adapted to a closed system with an electric heater (approximately 80 kilowatt thermal [kWth]) with an outlet temperature of 1300ºF (730ºC). Improved heater systems that better simulate the thermal hydraulics of nuclear reactors and that are capable of providing higher temperatures and more power can be used in the future. At the present time the heater is limited to 63 kW and 1200ºF (630ºC) outlet temperatures. The Figure A.9.6.1-1 shows an engineering drawing of the Brayton loop and an actual photo as installed at Sandia with the insulation added to the loop.

Number of employees in facility:

The Brayton Facility requires only a single trained operator to run most experiments.

Types and quantities of radioactive wastes generated:

There is no routine waste generation at the facility. When alternate working fluids are used in experiments (CO_2, inert gases, or mixtures) the gases are generally vented when the series is completed. Working fluids used to date have not required special environmental safety and health procedures. There are no radioactive materials or radiological issues with this facility. Small quantities of hazardous materials are used in general cleaning and maintenance activities.

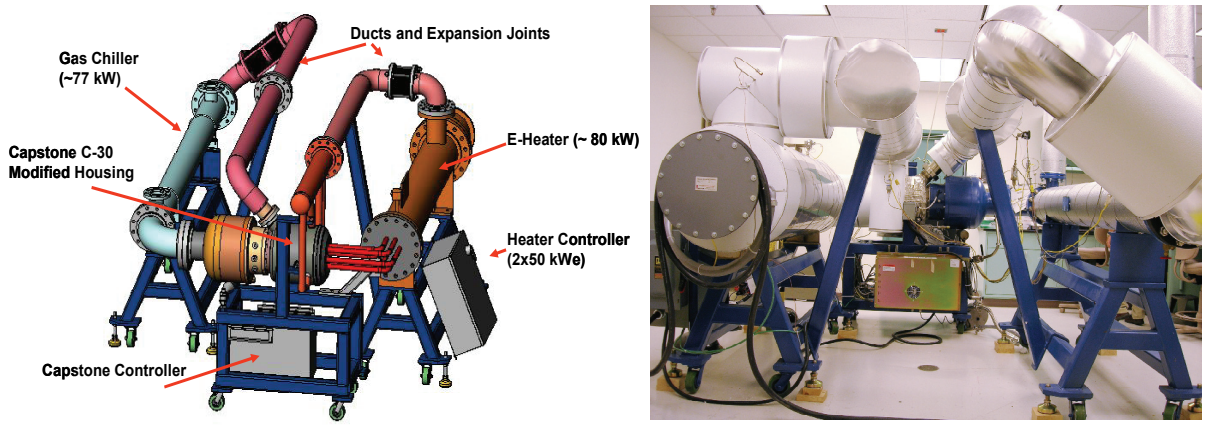
These materials are collected and managed on-site. Hazardous and LLW at SNL/NM is processed at 5 facilities: the, the Thermal Treatment Facility (TTF), the HWMF, the RMWMF, the Manzano Storage Bunkers (MSB), and the Solid Waste Transfer Facility (SWTF).

Significant radionuclide emissions:

There are no routine radiological emissions during operation.

Average person-rem for worker in facility:

The facility does not produce any worker exposure.



Source: Burns 2008

FIGURE A.9.6.1-1—*Engineering Layout Drawing of Spent Nuclear Fuel Closed Brayton Loop (left) and Picture of Brayton Loop with Full Insulation*

Information on age of facility and plan for future use:

The current closed Brayton loop would continue to be used as a test bed for investigation of control strategies, alternate working fluids and coupled reactor dynamics studies. The current priority for Brayton cycle studies is on the supercritical CO_2 power conversion system for sodium or other intermediate temperature reactor systems. A series of S-CO_2 test loops are being developed under the Generation IV program to provide the capability for investigation of supercritical CO_2 systems at the megawatt level. The first unit would be a single compressor loop to study compression near the critical point of CO_2. Additional components for split flow compressor tests and power producing S-CO_2 Brayton systems would be added to allow full investigation of the S-CO_2 cycle for advanced reactors. The first S-CO_2 compressor tests were conducted during fiscal year 2008 with full Brayton cycle capabilities available in fiscal year 2009.

A.9.7 Savannah River National Laboratory

The Savannah River National Laboratory (SRNL) supports a wide range of AFCI research including activities associated with advanced separations, safeguards, and waste form development. Most AFCI research is performed at the laboratory's 773-A building, with some

non-radioactive process development work performed at the Aiken County Technology Laboratory. The SRNL is part of DOE's Savannah River Site (SRS) which operates the country's only full scale reprocessing facility and has operating support facilities including a broad spectrum of waste management operations, SNF receipt and storage capabilities, and actinide stabilization and storage processes. These facilities have available laboratory space and could provide additional support to the AFCI program in the future.

A.9.7.1 *Building 773-A*

Basic description of facility and type of work performed:

SRNL's main laboratory building (773-A) is comprised of approximately 290,000 ft^2 (27,000 m^2) of space. The facility contains both radiological and non-radiological laboratory modules.

Source: Burns 2008

FIGURE A.9.7.1-1—*The Savannah River National Laboratory*

The Shielded Cells Facility within 773-A contains 16 cells that are designed to remotely handle material with doses as high as 10,000 rem/hour. The cells provide 576 ft^2 (54 m^2) of cell footprint. The Shielded Cells were used for a demonstration of the UREX process in 2002 and would be used when high dose rate materials are handled for the AFCI R&D, such as demonstration of the separations processes and waste form development.

Source: Burns 2008

FIGURE A.9.7.1-2—*The Shielded Cells Facility at the Savannah River National Laboratory*

773-A also has the Intermediate Level Cells (ILC), consisting of two cells totaling 72 ft^2 (7 m^2) of cell floor space, designed to shield the radiation from a 5 Ci Cs^{137} source. These ILCs were used for experiments involving the decontamination of the zircaloy hulls resulting after dissolution of the reactor fuel. Future experiments involving fuel hulls and undissolved solids (UDS) could be performed in the ILC.

A.9.7.2 *Aiken County Technology Laboratory*

Basic description of facility and type of work performed:

The Aiken County Technology Laboratory (ACTL) is a laboratory facility leased by SRNL at the Savannah River Research Campus. The ACTL provides 21,000 ft^2 (2,000 m^2) of laboratories and is designated for non-radioactive work exclusively. R&D work at ACTL that is in support of the AFCI would involve cold chemical demonstration of equipment and processes prior to deployment in the radioactive environments at Building 773-A, specifically the separations and waste development R&D efforts.

Number of employees in the facilities:

Approximately 700 of the 900 SRNL employees at the ACTL are directly linked to R&D work.

Types and quantities of radioactive wastes generated:

SRNL generates radioactive waste from its laboratory R&D and operational activities. The types of waste generated include sanitary waste, hazardous waste, LLW, transuranic waste, chemical substances listed under the *Toxic Substances Control Act* (TSCA) (15 U.S.C. 2601 et seq.) and mixed LLW. The wastes are typical of laboratory research and analytical processes. Radioactive wastes generated at SRNL are generally processed and/or packaged in waste management areas located within the 700 Area. DOE uses a number of methods for treating and disposing of LLW at SRS, depending on the waste form and activity. Some LLW that is not technically or economically suitable for disposal at SRS is shipped to the NTS or the EnergySolutions Disposal Facility in Utah, for disposal. At present, DOE stores hazardous wastes in three buildings and on three pads that have RCRA permits. SRS hazardous waste streams consist of a variety of materials, including mercury, chromate, lead, paint solvents, and various laboratory equipment. Hazardous waste is sent to offsite treatment and disposal facilities. DOE TRU waste is to be disposed of at the WIPP in New Mexico. SRS stores transuranic waste from past DOE onsite and offsite operations on concrete pads. In 2001, SRS initiated its program to re-package TRU waste and ship it to the WIPP.

TABLE A.9.7.2-1—*Fiscal Year 2008 Forecasted Volume of Radioactive Waste*

Waste Category	Fiscal Year 2008 Forecast (m^3)
Sanitary Waste	20.4
Hazardous Waste	4.48
Low Level Waste	390
Transuranic Waste	7.64
TSCA Waste	0.12
Mixed Low Level Waste	6.92

With the AFCI R&D work currently representing only approximately 1 percent of the SRNL R&D budget, it can be concluded that the AFCI work has a small impact on the existing waste volumes. All wastes are managed in accordance with applicable requirements.

Significant radionuclide emissions:

The SRNL contribution to the SRS's overall air emissions has been historically less than 1 percent. The radionuclide fraction from SRNL has historically been comparably small. With the AFCI R&D work currently representing only approximately 1 percent of the SRNL R&D budget, it is a credible conclusion that no significant emissions are produced by AFCI activities within the SRNL 700 Area.

Average person-rem for worker in facility:

The average dose to SRS personnel who were monitored in 2007 was 15 mrem/year. The average dose to the SRNL personnel who were monitored in 2007 was 5 mrem/year.

With the AFCI R&D work currently representing only approximately 1 percent of the SRNL R&D budget, it can be concluded that the AFCI work has a small impact on the existing dose rates.

Information on age of facility and plan for future use:

Building 773-A (and its associated buildings) was built in the 1950s as part of the development of the SRS. Upgrades and replacements throughout their history have kept them serviceable. They are in compliance with all regulatory requirements and have a current approved safety basis.

Future plans are to continue to maintain and use the 773-A facilities for the foreseeable future. ACTL opened in 2001. SRNL has a long-term ACTL lease with Aiken County.

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