MODULE 6.0: BACK-END OF THE FUEL CYCLE: SPENT NUCLEAR FUEL AND IRRADIATED MATERIALS

Introduction

Welcome to Module 6.0 of the Fuel Cycle Processes Directed Self-Study Course! This is the sixth of nine modules available in this directed self-study course. The purpose of this module is to provide an understanding of nuclear fuel and materials after their use for generating power in a reactor - referred to as the "back-end" of the fuel cycle - and the differences between the open and closed fuel cycles. Unlike the other parts of the fuel cycle, the spent nuclear fuel (SNF) is highly radioactive and heat-generating; and thus, the facilities, hazards, and safety approaches are fundamentally different. The module has seven learning objectives. This self-study module is designed to assist you in accomplishing the learning objectives listed at the beginning of the module. The module's self-check questions will help you assess your understanding of the concepts presented.

Before you Begin

It is recommended that you have access to the following materials:

- Trainee Guide

Complete the following prerequisites:

- Module 1.0: Overview of the Nuclear Fuel Cycle
- Module 5.0: Fuel Fabrication

How to Complete this Module

1. Review the learning objectives.
2. Read each section within the module in sequential order.
3. Complete the self-check questions and activities within this module
4. Check off the tracking form as you complete the self-check questions and/or activity within the module.
5. Contact your administrator as prompted for a progress review meeting.
6. Contact your administrator as prompted for any additional materials and/or specific assignments.
7. Complete all assignments related to this module. If no other materials or assignments are given to you by your administrator, you have completed this module.
8. Ensure that you and your administrator have dated and initialed your progress on the tracking form.
9. Go to the next assigned module.

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MODULE 6.0: BACK-END OF THE FUEL CYCLE: SPENT NUCLEAR FUEL AND IRRADIATED MATERIALS

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MODULE 6.0: BACK-END OF THE FUEL CYCLE: SPENT NUCLEAR FUEL AND IRRADIATED MATERIALS

LEARNING OBJECTIVES

6.1 On completion of this module, you will be able to identify and describe the properties of spent nuclear fuel (SNF), storage for spent nuclear fuel, reprocessing and recycle of spent nuclear fuel, high level waste (HLW), the SNF/HLW repository, and ongoing programs/NRC involvement.

6.1.1 Identify the properties of spent nuclear fuel.

6.1.2 Identify storage for spent nuclear fuel.

6.1.3 Identify reprocessing and recycle of spent nuclear fuel.

6.1.4 Identify high level waste.

6.1.5 Identify the repository for spent nuclear fuel and/or high level waste (Yucca Mountain).

6.1.6 Identify ongoing programs and NRC involvement with SNF.

OVERVIEW OF BACK-END OF THE FUEL CYCLE

The “back-end” of the fuel cycle, i.e., post-reactor use of the (now) irradiated fuel, addresses the management and disposal of spent nuclear fuel (SNF) and related radioactive wastes. The handling of SNF and its related radioactive wastes is a problematic, complex and controversial aspect of the nuclear fuel cycle today.

When a nuclear reactor operates in a thermal neutron spectrum (i.e., moderated, such as with water in a PWR or BWR), fissile uranium (uranium-235 in current power reactors) is partially consumed by fission and absorption reactions; the fission reactions create energy, more neutrons to continue the fission chain reactions and fission products, while the absorption reactions generate transuranic elements (TRU), primarily isotopes of plutonium, neptunium, americium, and curium. Fission products also absorb neutrons. The enrichment level of the uranium fuel compensates for neutron losses and absorption, and the enrichment decreases as uranium-235 is consumed. Ultimately, the decrease in enrichment and accumulation of fission

products and TRU reach a point at which the fission reactions cannot be sustained for the enrichment level, and the reactor has to be refueled - for PWRs and BWRs, refueling requires shutdown. At this point, the used fuel is said to be "spent" and must be replaced. However, the accumulated fission products and TRU isotopes continue to emit very high levels of radiation and heat, and pose difficulties in management and disposal.

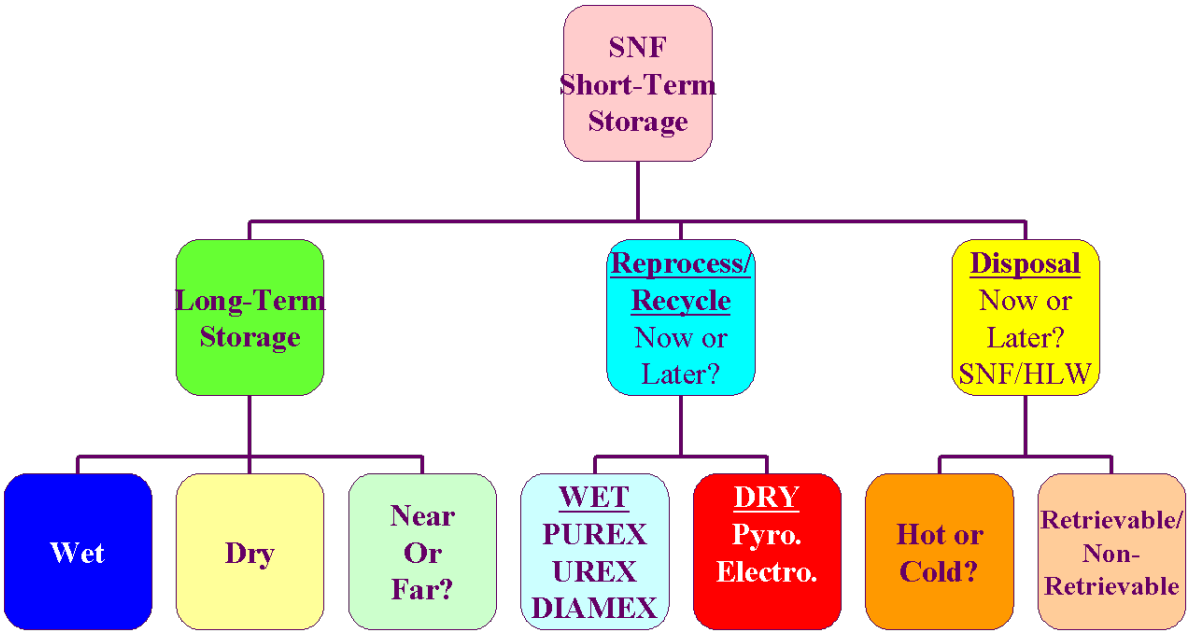
Figure 6-1 shows an SNF assembly being moved during refueling. The blue glow arises from Cerenkov radiation. A layer of water (usually at least 20 feet deep) shields personnel from the intense radiation levels and also provides cooling. Typically one-third of the core (about 33 MTHM; for a typical PWR, perhaps 66 assemblies) is removed and replaced with fresh fuel every refueling outage. Approximately 2,000-2,500 MTIHM of SNF are generated annually and about 50,000 MTIHM of SNF are currently stored at the reactor sites (2005 data).

Figure 6-1. SNF Assembly Being Removed During Refueling

Spent nuclear fuel (SNF) typically contains 94-96% uranium, about 1-1.3% plutonium, 3-5% fission products, traces (0.2% or so) of other transuranic compounds, and stable (non-radioactive) fission products. Thus, unlike the ash from fossil fuels, SNF has considerable potential energy content and valuable constituents - only 3 - 5% is actually "ash." SNF is extremely radioactive - in the absence of shielding, lethal radiation fields exist. SNF generates heat from the decay of the radionuclides. More information on SNF properties can be found in Section 6.1.1.

As shown in Figure 6-2, there are three main options for SNF: storage, reprocessing/recycle, or disposal. Each option has several sub-options. These are briefly discussed below and in their respective sections.

Figure 6-2. SNF Options and Decision Tree



All SNF from U.S. reactors is currently stored at or adjacent to reactor sites. SNF is initially discharged and stored under water in a spent fuel storage pool, hence the term "wet storage." After wet storage for a minimum of five to ten years (i.e., to allow for decay of both radioactivity and heat generation), the intent of most U.S. utilities is to transfer the SNF to dry storage in containers back-filled with helium. Natural circulation airflow around the outside of the containers provides the actual cooling. Many reactor sites have dry storage facilities, and more are planned. SNF storage is discussed further in Section 6.1.2.

Reprocessing is the term given to the separation and recovery of the 94-97% of the SNF that has remaining potential energy and value. This is accomplished in heavily-shielded, remotely operated and maintained facilities. Historically, the uranium and plutonium are recycled to reactors (such as in MOX fuel) while the other constituents, primarily fission products and some transuranics, are treated as waste, usually by vitrification. The recovered uranium and plutonium have relatively low radiation fields and can usually be contact-handled (i.e., unshielded gloveboxes) for fuel fabrication or storage. Section 6.1.3 discusses reprocessing and recyling in more detail.

High level waste (HLW) is the term applied to the highly radioactive fission product wastes from reprocessing. HLW contains the majority of the isotopes generating the intense radiation and heat and is usually generated in a liquid form. After storage, it is usually vitrified with glass forming reagents and stored in sealed stainless steel containers, pending subsequent disposal

in a repository. See Section 6.1.4 for a discussion of HLW. However, from a regulatory perspective, SNF is designated as HLW even though it is generated as a solid.

Performance assessments indicate that long-term isolation of the SNF and/or HLW from the environment is necessary for acceptable, future radiation doses to the public. A disposal site offering such long-term isolation is referred to as a repository. In the U.S., the proposed repository for SNF and HLW is Yucca Mountain in Nevada. This is discussed further in Section 6.1.5.

Currently, the NRC is heavily involved as the regulator for SNF storage and the proposed SNF/HLW repository for Yucca Mountain. DOE has stated that a combination of government and commercial facilities would be utilized in the Global Nuclear Energy Partnership (GNEP) program, and that GNEP facilities should be NRC-licensable. These programs and the NRC's involvement are discussed in Section 6.1.6.

MODULE 6.0: BACK-END OF THE FUEL CYCLE: SPENT NUCLEAR FUEL AND IRRADIATED MATERIALS

Learning Objective

When you finish this section, you will be able to:

6.1.1 Identify the properties of spent nuclear fuel.

6.1.2 Identify storage for spent nuclear fuel.

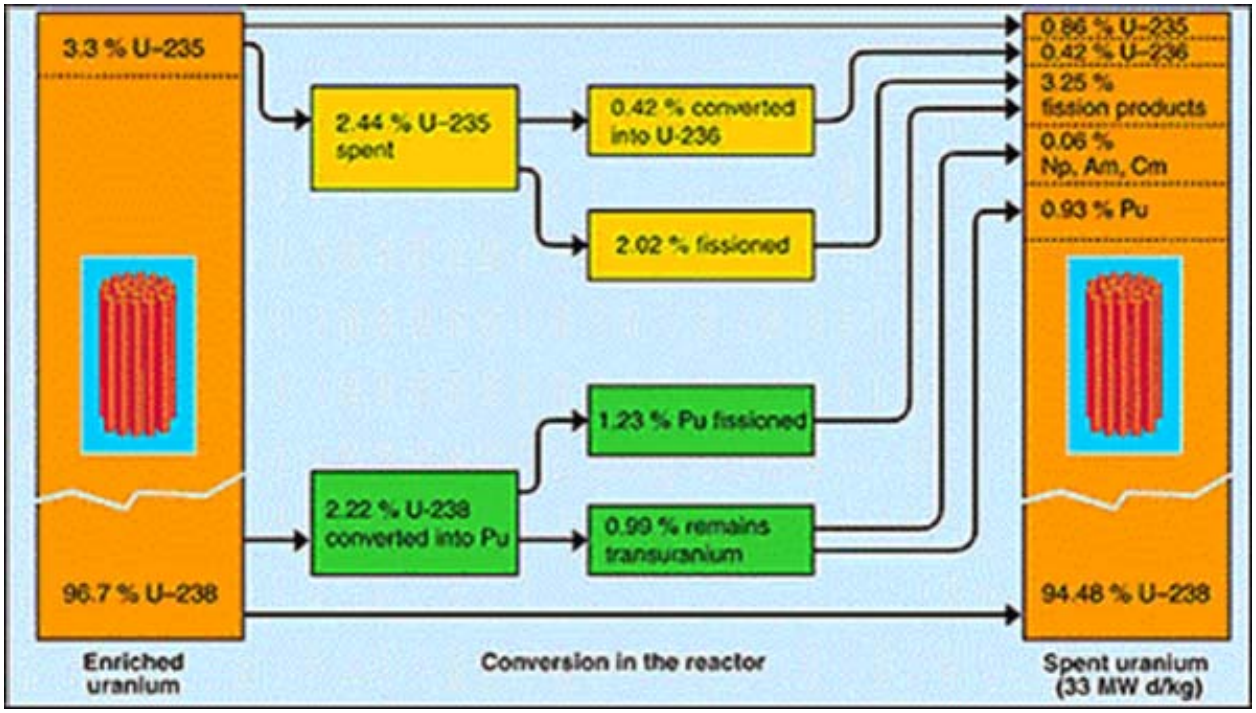
SNF PROPERTIES

The fissioning, or burning, of uranium fuel in the reactor core produces fission products and TRU. In general, SNF contains the following main categories:

- unused uranium
- plutonium, primarily the 238, 239, 240, 241, and 242 isotopes; this is sometimes referred to as reactor plutonium due to the presence of significant percentages of isotopes other than plutonium-239
- other TRU, such as neptunium, americium, and curium
- radioactive cesium (Cs/Ba) and strontium (Sr/Y) fission products (half-lives around 30 years)
- other long-lived radioactive fission products, with relatively long half-lives (> five years; usually > 100 years, such as technetium and iodine)
- other radioactive fission products, with relatively short half-lives (< five years, such as ruthenium)
- non or slightly radioactive fission products, such as xenon and the platinum group metals

Figure 6-3 provides a top-level estimate for low-enriched uranium SNF corresponding to about a one year refueling interval and a three year irradiation time in the core (i.e., a burnup of approximately 30,000 MWD/MTIHM; burnup is defined in the next section). Most of the uranium remains, although at a lower enrichment level (typically 1-1.5% assay if starting at 4.5-5% assay in the fresh fuel). A small percentage of the uranium-235 that absorbs a neutron forms uranium-236, which is itself a neutron absorber; a subsequent neutron absorption and beta decay yield neptunium-237.

Figure 6-3. Example of SNF Constituents



Note that some of the plutonium created with neutron absorption by uranium-238 fissions and contributes to the power of the reactor. At shutdown, around 40-50% of the power is derived from plutonium fission. Successive neutron absorptions by uranium and plutonium form the other TRU isotopes. Most TRU have relatively long half-lives; for example, about 24,000 years for plutonium-239 and 2.2 million years for neptunium-237. Shorter-lived TRU, such as americium and curium, contribute to the external radiation fields of TRU materials. All TRU isotopes are significant inhalation hazards and require careful confinement, such as gloveboxes and HEPA filter systems.

As illustrated by Figure 6-4, a spectrum of fission products are produced. There are two peaks - one roughly corresponding to strontium/yttrium and the other corresponding to cesium/barium. The distribution changes slightly, depending on the isotope undergoing fission. Radioactive cesium-137 and strontium-90 constitute the principal isotopes usually formed. Each has a half-life around 30 years and is in equilibrium with a short-lived daughter isotope. Both contribute to the heat load of SNF and cesium contributes significantly to the radiation fields because of its 662 KEV gamma emission. Long-lived fission products, such as technetium and iodine, generally do not present immediate handling hazards but usually contribute significantly to ingestion dose estimates for SNF disposal. Short-lived fission products (e.g., ruthenium-106) with typical half-lives of 6-18 months generate heat and significant radiation

fields, and can impact SNF reprocessing. Figure 6-5 shows that fission product activity decays rapidly with time. Figure 6-6 displays an isotopic breakdown of SNF decay on logarithmic coordinates. Again, the activity decays rapidly but there are multiple inflections in the total activity curve approximately corresponding to 10 times the half-life for short-lived isotopes, cesium/strontium, americium, and the long-lived fission product technetium-99. Significantly, the SNF activity falls below the activity level of the original uranium ore in the 700-1,000 year range.

Figure 6-4. Distribution of Fission Products from Uranium-235

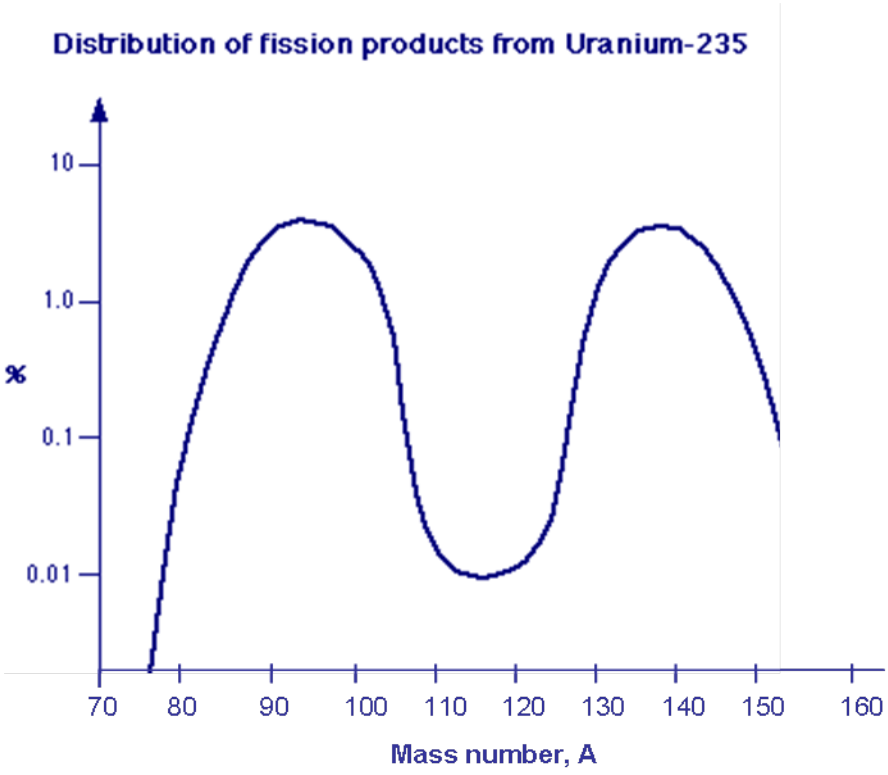


Figure 6-5. Radioactive Decay of Fission Products

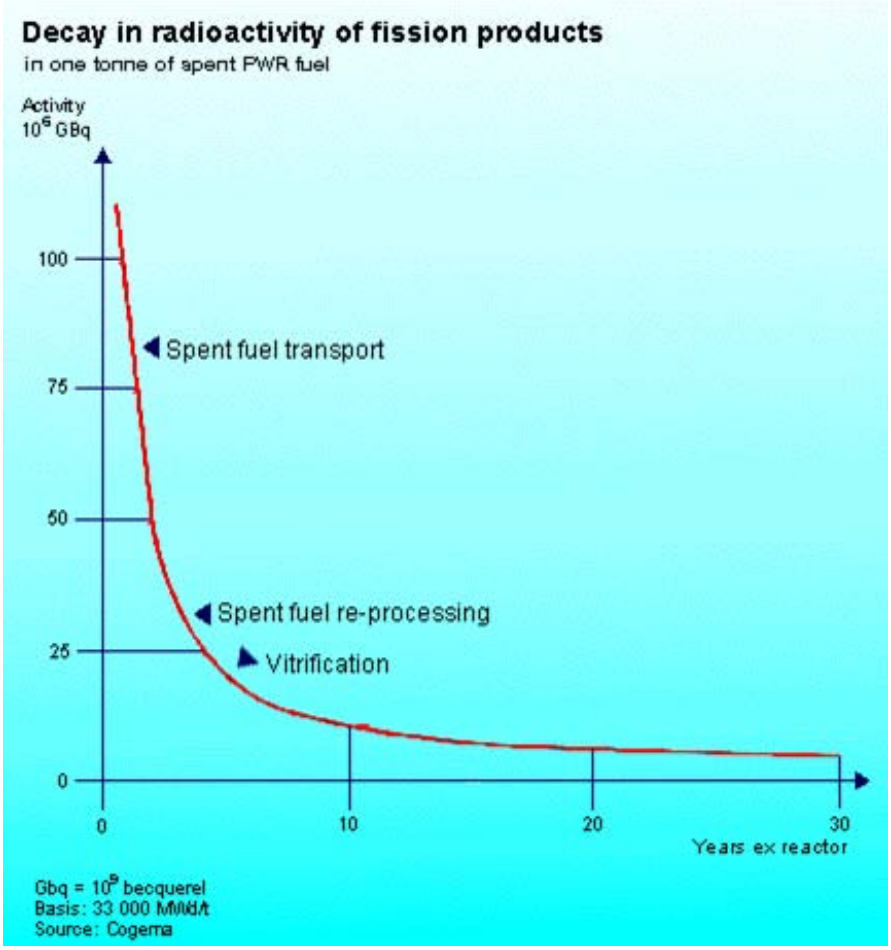
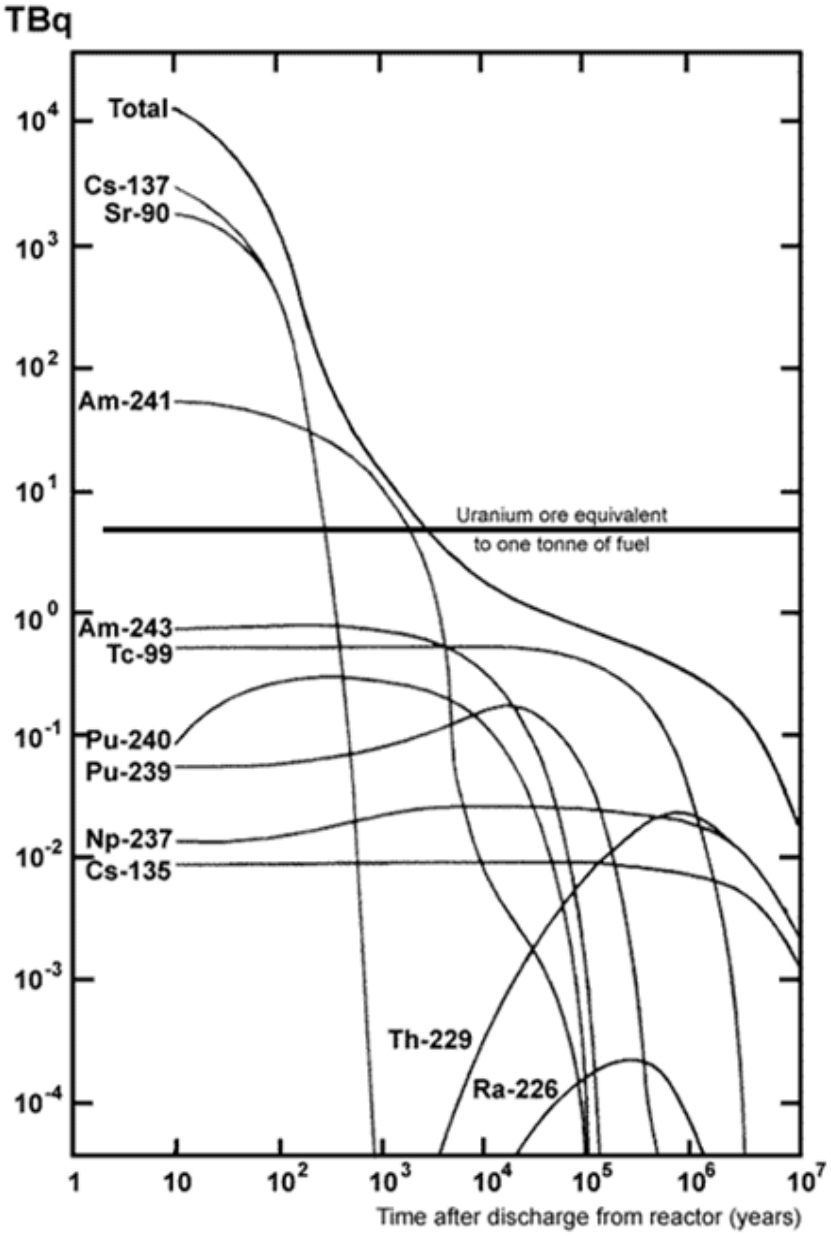


Figure 6-6. Isotopic Breakdown of SNF Radioactive Decay



Activity of high-level waste from one tonne of spent fuel

Source: IAEA, 1992 - radioactive waste management.

Cesium and strontium account for approximately 20% of the integrated heat load from the SNF, primarily in the first 300 years. The TRU isotopes (including plutonium) account for close to 80% of the integrated heat load from SNF.

Stable (non-radioactive) fission products include rare earth metals, members of the platinum group, and xenon. These are valuable materials, and, if recovered, could significantly defray the costs of reprocessing. However, they are not currently pursued for recovery.

While the civil nuclear fuel cycle generates SNF and its related wastes, these do not become "pollution" as they are contained and managed. In fact, nuclear power is the only energy-producing industry which takes full responsibility for its wastes and incorporates the charges into the overall economics of generating power. Furthermore, the expertise developed in managing these civil wastes is now starting to be applied to military nuclear wastes; the latter pose a real environmental problem in a few parts of the world.

Radioactive wastes from managing and processing SNF comprise a variety of materials requiring different types of management to protect the workers, the public, and the environment. In the United States, SNF and related wastes are normally categorized into high-level (HLW), transuranic (TRU), or low-level (LLW). SNF and HLW are defined in 10 CFR 60 and 10 CFR 63, based upon origin. TRU waste is defined in 10 CFR 61 based upon TRU concentration limits. LLW is also defined in 10 CFR 61 based upon concentration limits and is further divided into four categories; Class A (least radioactive), Class B (intermediate radioactivity), Class C (highly radioactive), and greater than Class C (GTCC - very highly radioactive). It is important to note that even Class A LLW can emit high radiation fields that can result in lethal doses in a short period of time. Other countries use similar limits and sometimes include an intermediate-level waste (ILW) for wastes between LLW and HLW that do not generate significant amounts of heat. A factor in managing wastes is the time they are likely to remain hazardous. This depends on the kinds of radioactive isotopes in them, and particularly the half lives characteristic of each of those isotopes. The half life is the time it takes for a given radioactive isotope to lose half of its radioactivity. After four half lives, the level of radioactivity is 1/16 of the original and after eight half lives, 1/256. In general, SNF/HLW, TRU, and GTCC are not suitable for near surface disposal and require geologic isolation. In contrast, LLW is suitable for near surface disposal. The NRC does conduct WIR (waste incidental to reprocessing) determinations for waste materials that by origin might be HLW but by hazard might be LLW; if the determination is positive, the waste material may be handled and dispositioned as LLW.

The various radioactive isotopes have half lives ranging from fractions to a second to minutes, hours or days, through to billions of years. Radioactivity decreases with time as these isotopes decay into stable, non-radioactive ones. The rate of decay of an isotope is inversely proportional to its half life; a short half life means that it decays rapidly. Hence, for each kind of

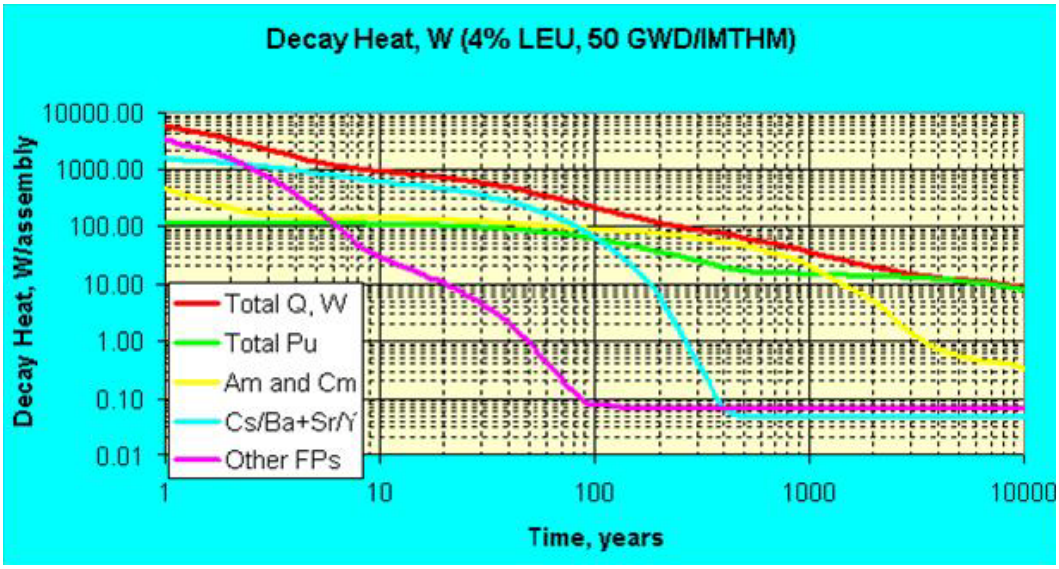
radiation, the higher the intensity of radioactivity in a given amount of material, the shorter the half lives involved. Three general principles are used in the management of radioactive wastes:

- concentrate and contain
- dilute and disperse
- delay and decay

The first two are also used in the management of non-radioactive wastes. The wastes are either concentrated and then isolated, or very small quantities are diluted to acceptable levels and then discharged to the environment. The third, delay and decay, however, is unique to radioactive waste management; it means that the waste is stored and its radioactivity is allowed to decrease naturally through decay of the radioisotopes in it. In the United States, the long-term storage of SNF and HLW has resulted in the decay of essentially all short-lived fission products and a significant reduction in the self-heating. Overseas, some countries wait several years to allow for the decay of short-lived fission products but reprocess five to seven years after reactor discharge to recover the energy value of plutonium-241, which has a relatively short half-life of 13.3 years.

SNF heat generation closely follows radioactive decay trends with one exception; americium ingrowth from the beta decay of plutonium-241. Figure 6-7 provides estimates of heat generation for an SNF assembly, broken down by major SNF isotope groups.

Figure 6-7. Typical SNF Assembly Heat Generation



Burn-Up

Burn-up represents the thermal energy recovered from the SNF and is a measure of the amount of fission that has occurred. Figure 6-8 shows actual data from the United States and Figure 6-9 summarizes world data. Typical values for U.S. reactors are in the 50,000-60,000 MWD/MTIHM range The trend in recent years is towards increased burn-up, with correspondingly higher initial uranium enrichment of the fuel assemblies.

Figure 6-8. SNF Burn-Up Trends in the United States

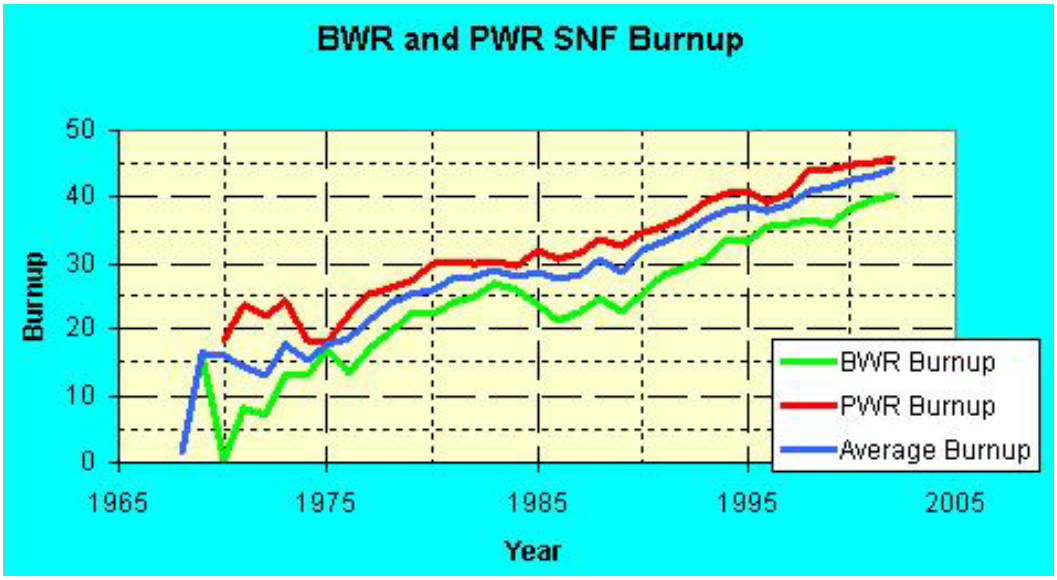
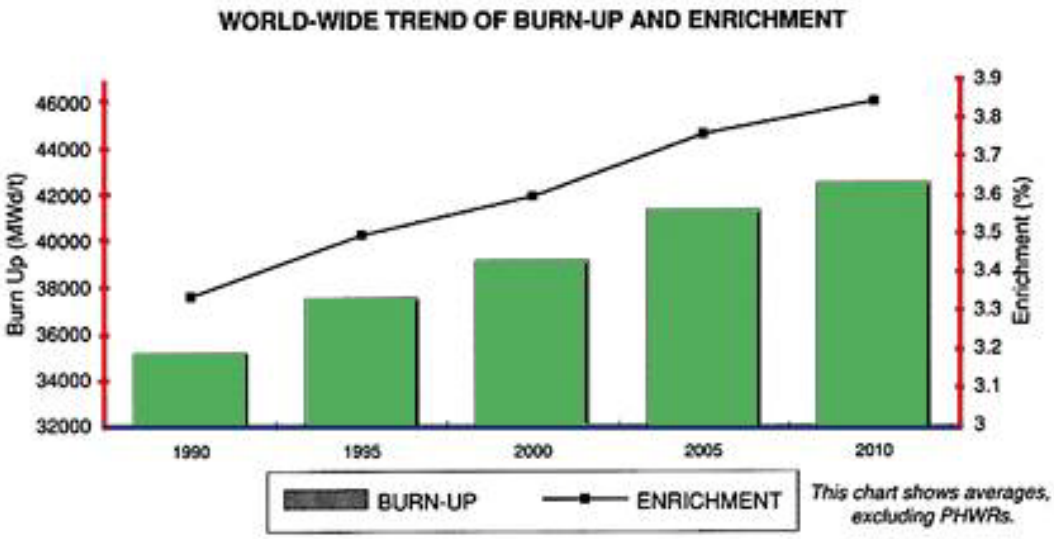


Figure 6-9. SNF Burn-Up Trends Worldwide

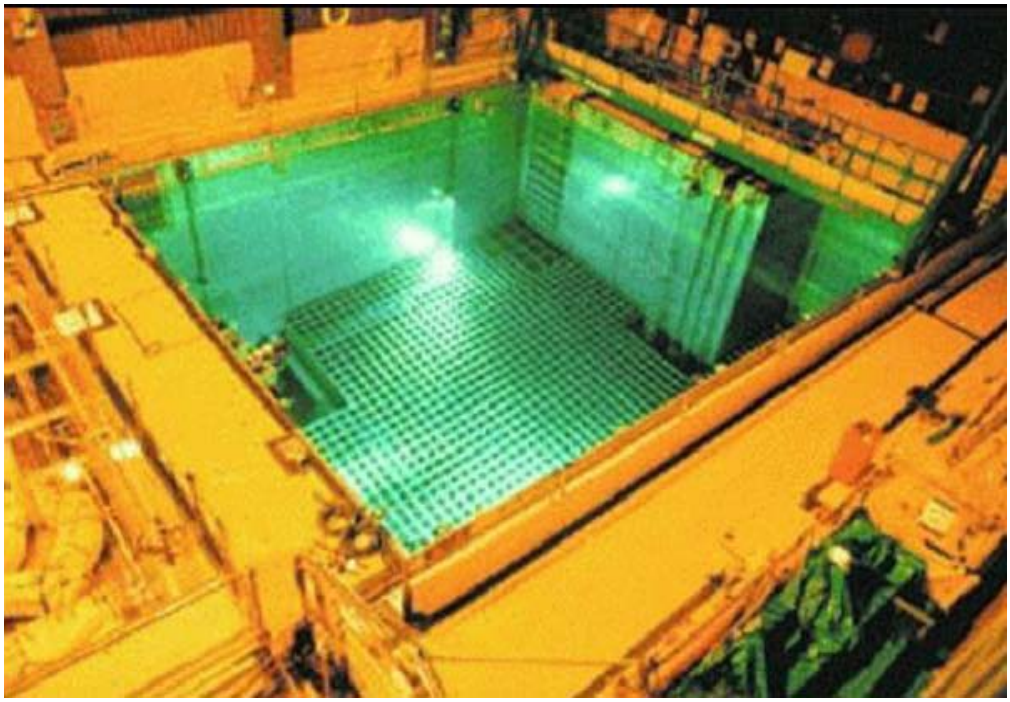


SNF STORAGE

Wet Storage of Spent Fuel

Approximately 2,000-2,500 MTIHM of SNF are generated annually in the U.S., and the U.S. reactor sites currently store about 50,000 MTIHM onsite (2005 data). These values seem large but are very small in perspective; the current inventory of SNF could be wet stored two assemblies high on an area the size of a single football field. SNF is removed from the reactor during the refueling shutdown. The SNF is removed from the core and transferred to the spent fuel pool (see Figure 6-10); all operations are conducted with the SNF underwater and using remote tools and equipment. Spent fuel assemblies that are removed from a reactor are very radioactive and produce heat. The spent fuel rods are initially stored in water which provides both cooling (the fuel generates decay heat as a result of residual radioactive decay) and shielding (to protect the workers from the intense radiation fields of SNF). The shielding is provided by a water depth covering the SNF (20+ ft. typical).

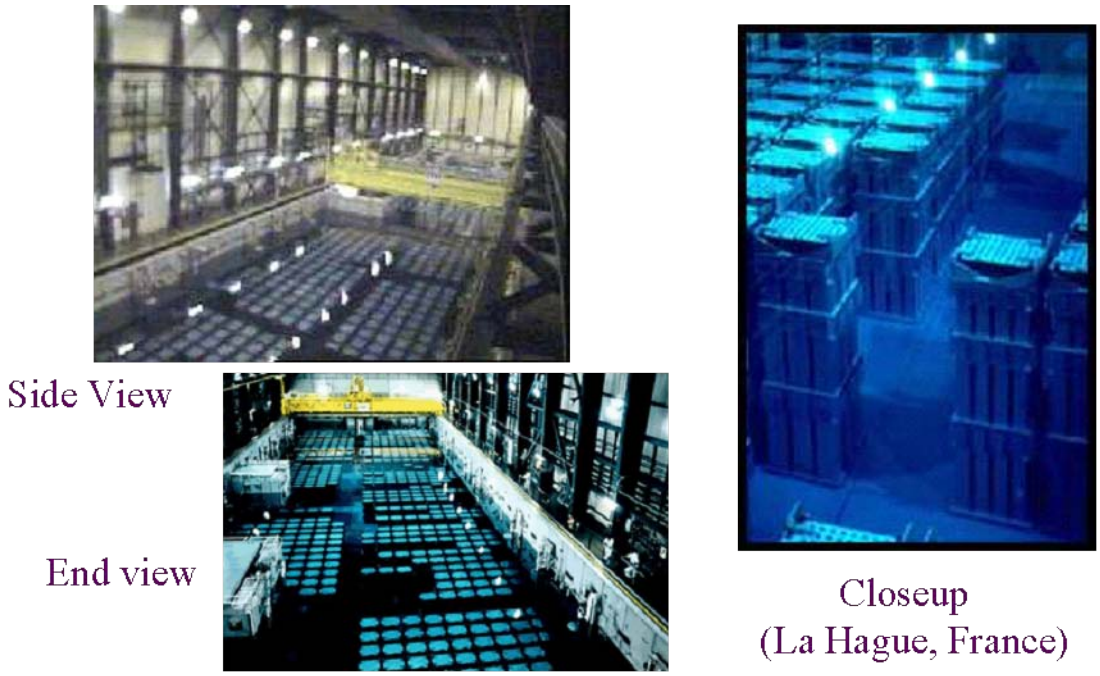
Figure 6-10. Typical SNF Storage Pool at a Reactor



Several health and safety concerns are associated with with storage pools include containment, sub-criticality, heat removal, radiation shielding, and active systems requiring active monitoring and maintenance. The effect of a complete loss of cooling water has been identified as the most severe potential accident scenario. Activities in the storage pool may cause mechanical damage to storage racks or to the pool in case of handling failures. However, only a few minor incidents have occurred. Wet storage occurs at low temperatures of 40-50 C (100-120 F). Wet storage at a reactor is licensed under 10 CFR Part 50 and contains relatively small quantities of SNF, perhaps 1,000-2,000 MTIHM per site. At the present time, the majority (close to 90%) of SNF in the United States is stored in reactor SNF storage pools.

There is one away from reactor wet storage facility licensed under 10 CFR Part 72 (Morris, IL). There are also larger, away from reactor wet storage facilities overseas associated with reprocessing plants that typically hold 3,000 MTIHM or more per pool (Figure 6-11); for example, the total wet storage capacity at the La Hague facility in France is approximately 13,000 MTIHM.

Figure 6-11. Large SNF Wet Storage Pools at La Hague, France



Dry Storage of Spent Fuel

After a period of cooling (minimum time - typically five years), fuel rods may be moved to dry cask storage. The actual time depends on the decay heat per assembly. The storage containers consist of three main parts:

- An inner metal basket that holds and secures the SNF assemblies
- An unshielded metal canister (container) that holds the basket and SNF, and isolates the SNF
- An outer container of either metal or concrete that provides for gamma and neutron shielding, and protection from external events and the environment

Dry storage can utilize the metal canisters in a horizontal position, inside a concrete vault, or in a vertical position, usually within an integral outer shielding container.

Overall, dry storage dimensions vary, but typical values are 5'-10' in diameter and 14' - 20' high. In a typical application, the metal canister is placed inside the SNF storage pool and loaded with SNF under water. The canister is partially raised into a shielded area and a top closure is welded in place (some designs use metal seals). The canister is further raised into the shielded container and water drained via small outlets. Vacuum drying is used to remove residual moisture and the canister is pressurized with helium. Operators weld the small openings closed and conduct leak tests. A truck transfers the loaded SNF canister and container/shield to the SNF dry storage area (usually a concrete pad or a vault). Equipment places the entire canister/container on the pad or, with horizontal vault designs, transfers just the SNF canister into the vault and reuses the transport container/shield.

Concrete container/shield systems require air entries and exits, with labyrinthine air cooling passages for cooling by natural convection. All metal systems can adequately transfer the SNF decay heat away by conduction and do not require air passages for cooling. However, metal container systems are considerably more expensive and the trend has been towards concrete containers for shielding.

The “high” temperature for dry storage is 200 - 300 C (typical) in pressurized helium, and 8-12" of steel or 3 - 4 ft of concrete is used for shielding. The concrete requires air passages for cooling, and seals and passages require routine inspections (Figure 6-12). At some sites, modular dry storage facilities known as independent spent fuel storage installations (ISFSI) provide economical storage. Figure 6-13 shows an example of horizontal storage modules.

Figure 6-12. Vertical Dry Storage Showing Air Inlets and Outlets

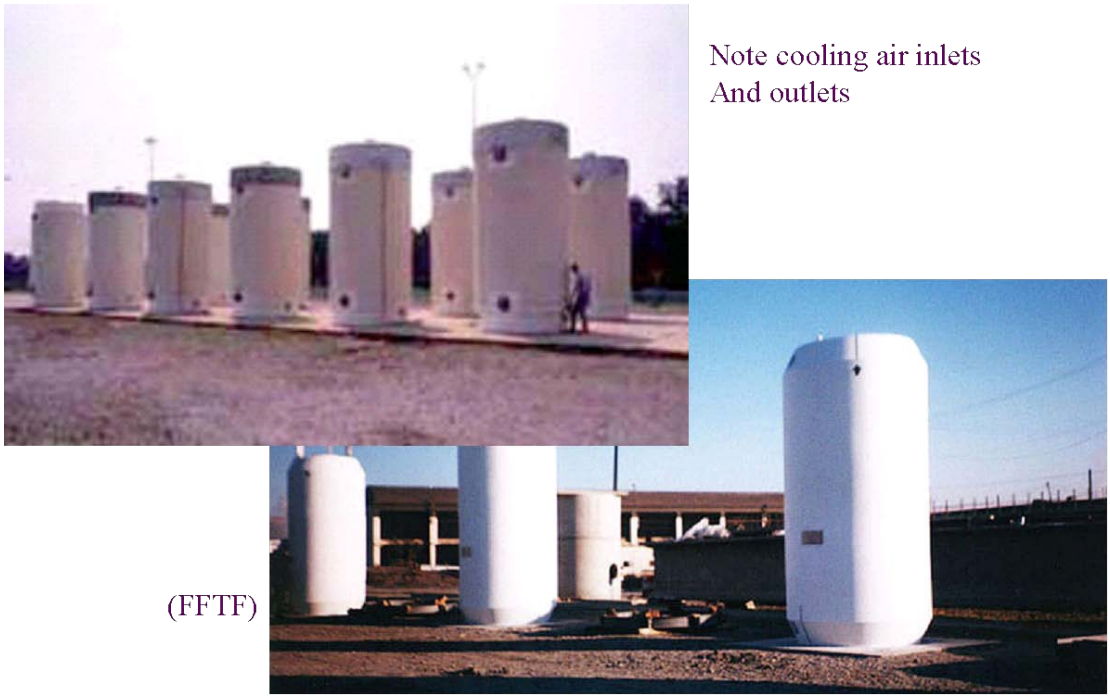


Figure 6-13. Horizontal Modules for Dry Storage of SNF

Dry storage appears to be attractive not only from an economic standpoint, but from its significant safety attributes, which are:

- inherently safe cooling by natural air convection
- sub-criticality without moderation of the fuel
- no necessity for permanent water treatment and no discharges of radioactive substances into the environment
- robustness against external impact and strong shielding by solid materials
- generally passive systems requiring little maintenance and inspection

The use of dry storage is growing, especially for aged fuel. (Dry storage is also used for HLW.) However, issues with dry storage include: metallurgical creep of cladding, burnup/credits, thermal problems, life extension (20+ years), transportation (of casks), storage away from reactor, and repository acceptability. Dry storage is licensed under10 CFR Part 72.

At some point, all currently loaded dry SNF storage casks may need to be unloaded for placement in an SNF repository. SNF disposal planning anticipates a facility at Yucca Mountain, Nevada, by 2017 for permanent disposal of the spent assemblies. See Section 6.1.5.

Self-Check Questions 6-1

INSTRUCTIONS: Fill in the missing words in each statement. Answers are located in the answer key section of the Trainee Guide. Choose from the following words.

Complete the following questions. Answers are located in the answer key section of the Trainee Guide.

1. What are the three principal options for managing SNF?

2. What percentage of uranium typically remains in SNF, and what is its approximate assay?

3. Name the four TRU isotopes usually found in SNF.

4. Describe the two main hazards of SNF and HLW.

5. Name the two main fission products produced - the “peaks” on the fission product distribution curve.

6. Identify typical burnup values for U.S. SNF.

7. What is the approximate quantity of SNF currently in storage at nuclear power plants, and what is the average annual increase?

You have completed this section.
Please check off your progress on the tracking form.
Go to the next section.

Learning Objective

When you finish this section, you will be able to:

6.1.3 Identify reprocessing and recycle of spent nuclear fuel.

SNF REPROCESSING

Reprocessing refers to the breakdown of the SNF, processing, and separation and recovery of useful constituents. Treatment and solidification of the resultant waste streams are usually considered part of reprocessing. Reprocessing may utilize aqueous-based processes or non-aqueous processes - these are usually referred to as wet and dry reprocessing, respectively, although dry reprocessing may utilize a liquid phase.

Unlike the ash from fossil fuel power plants, SNF contains 94-96% uranium, 1% or so plutonium, and some TRU. All of these materials can be reused as nuclear fuel or treated as waste. Note that uranium is considered to be a LLW by NRC regulations and decisions, and, thus, reprocessing by itself may considerably reduce the quantity of materials categorized as HLW and requiring geologic disposal in a repository, as compared to direct disposal of SNF.

Reprocessing and recycle avoid the waste of a valuable resource. The uranium in SNF is LEU, typically with assays between 1-1.5% uranium-235, well above the natural assay of 0.71% even when the reactivity losses due to the small amounts of uranium-236 are included. This reprocessed uranium can be re-enriched for reuse in fresh, LEU nuclear fuel. This reduces uranium consumption and the need for additional uranium mining by perhaps 30%. Plutonium is also a fissile material with a significant energy content. It can be combined with depleted or uranium with other assay levels to manufacture mixed oxide fuel for energy production. The other TRU isotopes and uranium-238 are only fissionable and capable of generating energy in a fast (unmoderated) neutron spectrum. All current power reactors are moderated and, thus, cannot utilize TRU and uranium-238 as fuel.

The world's current reprocessing capacity is around 5,000 MTIHM/yr for commercial SNF (i.e., power reactor burn-ups or 30,000 MWD/MTIHM or more). This is 40% or so of the world's SNF generation rate of circa 14,000 MTIHM/yr. The largest commercial SNF reprocessing facilities correspond to about 800 MTHM/yr capacity each and exist in France (two at La Hague) and England (one at Sellafield). A new facility is being commissioned in Japan that will add approximately 800 MTHM/yr reprocessing capacity. As of 2002 (latest data), more than 80,000

MTIHM of SNF from commercial power reactors had been reprocessed. For comparison, several hundred thousand tonnes of low burn-up metallic SNF have been reprocessed for military purposes.

Most U.S. nuclear reactors operate on cycles longer than one year, and often refuel every 18 months. For a large reactor (nominally a 1,000 MWe LWR), about one-third of the core is removed each refueling, generating about 33 MTIHM of SNF. On an annual basis, this might be 22 or so MTIHM. This may also be estimated from the number of operating nuclear reactors (103 in the U.S.) and the known annual SNF generation rate (2,000-2,500 MTIHM/yr). A reasonable approximation is 25 MTIHM/yr for a nominal 1,000 MWe LWR. Currently, no reprocessing is performed in the U.S. A reprocessing capacity equivalent to France and England combined would be necessary just to address the current annual SNF generation rate in the U.S. from existing power reactors. Processing the current inventory would require additional capacity.

Reprocessing may be conducted anytime after SNF discharge from a reactor. Nuclear engineering and chemistry "textbooks" generally mention short times of 0.5-1 year. However, the radioactivity, number of radionuclides, and heat generation of SNF decrease rapidly after the first few years from discharge and this makes reprocessing and the safety basis easier. Conversely, commercial reuse of plutonium as MOX fuel is more economical in LWRs if the energy content of plutonium-241 is also recovered and americium-241 ingrowth avoided. Plutonium-241 has a 13.3 year half-life. Commercial reprocessing usually compromises, with reprocessing at approximately 5 years after discharge and reactor recycle as MOX after about 7 years after discharge.

Reprocessing involves the following process steps:

- Mechanical disassembly of the SNF
- Mechanical destruction of the SNF fuel rods ("chopping") into one-inch long segments
- Conditioning of the (now exposed) fuel material; for oxide fuels, this is usually a low temperature (200-400 C) oxidation of the dioxide to the trioxide or octaoxide. This causes expansion and porosity increases in the fuel material, releases volatile fission products, and eases subsequent dissolution.
- Dissolution of the fuel material
- Separation of the uranium and plutonium from the fission products and other TRU
- Separation of the uranium from the plutonium (performed in the current commercial plants)
- Conversion of the uranium and plutonium back into oxide forms

- Conditioning and solidification of the fission products and other TRU; vitrification is the principal solidification technology used at the current facilities
- Interim storage of SNF, uranium oxide (usually the trioxide), plutonium oxide (usually the dioxide), and solidified HLW

Reprocessing also involves numerous programmatic activities, including material control and accountability, that also address security and proliferation. In practice, reprocessing facilities usually have multiple barriers for containment of material and security, multiple interim storage areas/tanks for in-process materials, a seismically qualified basis, and thick walls and cells for shielding.

Wet Reprocessing Routes

Wet reprocessing routes utilize aqueous solutions and low, near ambient temperatures. Solvent extraction processes have been the most successful. Wet routes use concentrated nitric acid solutions to dissolve the fuel material. Fluoride salts and/or electrolysis may be used to increase dissolution kinetics. Once dissolved, additional chemicals adjust the pH and valence states; either hydrogen peroxide or hydrazine are added to bring the plutonium nitrate into a +4 state (the most extractible). Then, the aqueous phase is contacted - mixed - with an immiscible organic phase containing an extractant. The uranium and plutonium are preferentially absorbed into the organic phase while the fission products and other TRU materials preferentially remain in the aqueous phase as HLW. The process coalesces and separates the organic phase. The aqueous phase is conditioned and vitrified with glass formers. The organic phase may be further purified or scrubbed by other materials and chemicals. Next, a dilute nitric acid stream containing a reductant contacts the organic phase; the reductant reduces the plutonium valence to the +3 state, which preferentially migrates to the aqueous phase while leaving the uranium in the organic phase.

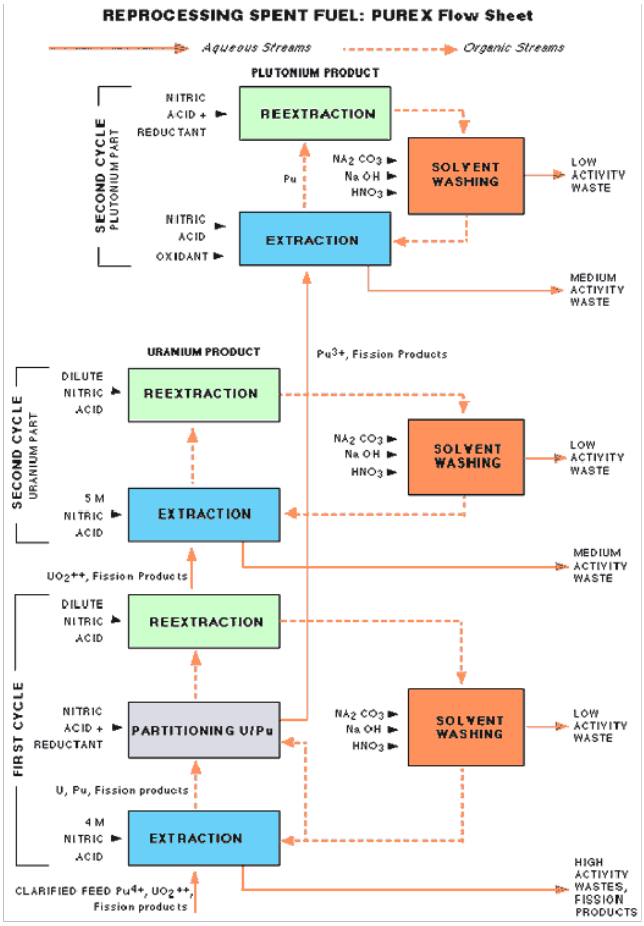
Potential reductants include ferrous sulfamate, hydrazine and hydroxylamine nitrate, and uranium(IV). A subsequent step contacts the organic phase containing the uranium with a different concentration nitric acid phase, which preferentially recovers uranium as the nitrate in the aqueous/acid phase. One solvent extraction cycle usually gives a one-thousand fold reduction in the fission product concentrations, i.e., a decontamination factor (DF) of 1,000. However, this is usually too radioactive for contact handling by workers, and a second cycle is normally applied to both the separated plutonium and uranium products. Contact handling normally requires a DF of one million or more and this is achieved by two complete solvent extraction cycles.

Many solvent extraction systems and variations exist. All large commercial facilities apply the Purex (Plutonium URanium EXtraction) process. Purex uses tributyl-phosphate (TBP) as the extractant in a kerosene solvent/diluent. Usually, hydrazine and hydroxylamine nitrate are the reducing agents. Purex achieves 99.8-99.9% recovery of uranium and plutonium, with a fission product DF of at least one million. The large commercial facilities in France and England have

economically applied the Purex process. The safety aspects of Purex are well developed but incidents do occur and accidents have happened in the past. Figure 6-14 displays a process flow diagram for the Purex process.

Other wet processes include Hexone/Redox, Butex, Urex, TRUEX, and others. These are solvent extraction processes similar to the Purex process but utilize different solvents and extractants.

Figure 6-14. Purex Process Flow Diagram



Dissolved Fuel Enters At Bottom Of Page

PUREX Process Block Diagram -Separates Pu from U

Dry Reprocessing Routes

Dry reprocessing routes are non-aqueous based, but liquids may still be involved. Sometimes called pyroreprocessing or pyrometallurgical, the dry routes mix melting, electrolysis, molten salts and/or volatilization to separate uranium and plutonium from fission products and other TRU isotopes.

Dry reprocessing includes:

- fluorination processes, which volatilize and recover the uranium
- organic distillation processes (e.g., ACAC - acetyl acetonate - like compounds)
- melt refining, which use melting of fuel and slagging to remove the fission products
- molten salt dissolution and electrolytic extraction, which separates the SNF constituents by electrochemical potentials

The latter has been studied extensively. At INL (Idaho), extensive pilot scale experiments have been conducted using spent metallic fuel from the EBR-II, as part of programs related to the Integral Fast Reactor (IFR) concept. SNF is placed in an anode basket or otherwise dissolved into a molten salt. Uranium and the actinides can be recovered at the cathode of an electrolysis process applied to the molten salt, and remanufactured into nuclear fuel. However, decontamination factors for dry processes tend to be significantly lower than for wet routes, and remote fuel fabrication would be needed. Electrolysis can also anodically dissolve the fuel. Fission products remain with the molten salt which requires occasional removal and replenishment. The removed salt can be converted into a ceramic waste form for HLW disposal. The process has been applied on a small scale and additional efforts would be needed to adapt the process to commercial oxide SNF and increase throughput to the levels required for commercial applications.

Reprocessing Facilities

Internationally, France, UK, India, Pakistan, Russia/FSU and China operate facilities to reprocess spent nuclear fuel. Japan has a pilot facility and is commissioning a full-scale reprocessing plant. The world's total capacity for reprocessing commercial SNF is around 5,000 MTIHM/yr.
In the U.S., reprocessing has been performed extensively with low burn-up fuels for defense purposes (over 100,000 MTIHM reprocessed, annual capacities of several thousand MTIHM annually). However, reprocessing of commercial fuel was limited to the West Valley Facility in New York State, with less than 1,000 MTIHM processed. A commercial plant built in Illinois did not operate due to technical problems, and a plant in South Carolina did not operate due to the domestic ban on reprocessing enacted by President Carter in 1977. Thus, commercial reprocessing has not taken place to any significant degree in the U.S. See Section 6.1.6 for DOE programs and future NRC involvement with U.S. reprocessing of SNF.

Reprocessing plants are large facilities. Figure 6-15 shows the Thermal Oxide Reprocessing Plant (THORP) in the UK. This commercial facility treats spent fuel from UK and overseas reactors, separating the high-level waste from uranium and plutonium. The nominal capacity is 800 MTIHM/yr.

Figure 6-15. THORP in the UK

The Thermal Oxide Reprocessing Plant (THORP) in UK. This commercial facility treats spent fuel from UK and overseas reactors, separating the high-level waste from uranium & plutonium. The smaller black building on the right is the vitrification plant for this waste.

NRC REGULATIONS FOR REPROCESSING FACILITIES

A reprocessing plant is considered a production facility and would be regulated under 10 CFR Part 50, the same regulation that is used for nuclear power plants. Requirements include accident analyses and identification of safety controls, applicable general design criteria (from Appendix A - not all apply), QA, fire protection, and seismic qualification. Appendix F contains specific requirements related to HLW management, including solidification and shipment to a Federal Repository.

Other operations at a reprocessing facility could be incorporated into the main Part 50 license or could also be licensed under other NRC regulations:

- Interim SNF Storage (front end of plant): Part 72
- Actinide Fuel Fabrication and SNM Storage: Part 70

- HLW Vitrification and Storage: Part 70 (contains SNM)
- Cesium, Strontium, and Other Waste Storage: Part 30 or 70
- Reprocessed uranium (RepU - likely LEU SNM)- Part 70

Part 51 contains the regulations for compliance with NEPA. The licensing of a reprocessing plant and its associated facilities would be considered a major Federal action and would require an environmental impact statement (EIS). In order to avoid sementation, the EIS would have to address all planned operations even if separate licenses are being pursued for schedule or business purposes.

REPROCESSING ENVIRONMENTAL, SAFETY, AND HEALTH CONCERNS, AND SAFETY APPROACHES

Some ES&H concerns are:

- Intense (and often lethal) radiation levels, and doses
- Self-heating materials
- Criticality, particularly for liquid phases and plutonium compounds
- Chemical reagents, for toxicity and reactivity, use of organic compounds in radiation fields, HVAC contradictions
- Fire and explosion
- Dissolution and insolubles, and fines
- Corrosion/erosion
- Environmental qualification of equipment
- HVAC approach, scrubbers, filters
- Waste treatment forms, qualification, solidification, and management/disposal

The general safety approach for reprocessing facilities starts with a robust building design, with multiple layers of confinement, seismic qualification, many remote operations, and thick shielding. Current commercial facilities use a labyrinthine cell approach instead of a canyon for hazard segmentation; the process piping and equipment are the first confinement boundary, a cell pan the second layer, the cell is the third layer, the cell/building area the fourth layer, and the building exterior is the fifth layer. Corrosion resistant materials are used throughout the facility - primarily stainless steels with more corrosion resistant materials for special operations (e.g., high nickel alloys and titanium for dissolvers).

The HVAC similarly follows a multiple confinement approach, with increasing negative pressures as the potential or actual contamination levels increase; the highest contaminated or potentially contaminated area is at the lowest (most negative) pressure, and, thus, radioactive materials are always drawn away from the environment. Multiple confinement layers of filters

remove contaminants from the HVAC exhaust, and, after final HEPA treatment, discharge is via an elevated stack. Filter layers usually include the cell/glovebox itself, the cell/building area, and two final HEPA filter banks. Sand filters may be used prior to stack discharge as another confinement layer. Venting of process vessels and equipment occurs via a similar, dedicated HVAC system that also includes scrubbers; one of the scrubbers is usually slightly alkaline for better removal of acid vapors and gases.

Waste management at the current reprocessing facilities in France and England was organized and developed prior to operations. HLW is vitrified at these facilities and stored in air cooled vaults. The current inventory contains about 8 gigacuries (for perspective, this is about an order of magnitude greater than the radioactivity in all of the DOE HLW). Cladding hulls are compacted and treated as intermediate waste (non-heat generating). Relatively small quantities of TRU waste are generated; most TRU isotopes are ultimately routed to the HLW. The facilities apply a waste equivalency rule - HLW is returned to the country of SNF origin equivalent to the sum of the actual HLW generated and the isotopic equivalent in the intermediate and LLW materials.

The commercial reprocessing experience of France and England currently achieves average radiation worker doses comparable to or below natural background. Maximum public individual doses are around a tenth of normal background. Incidents do occur at these facilities but, due to the robust construction and multiple layers of confinement, their impact is low.

SNF RECYCLE

The reprocessing of spent fuel generates multiple products; plutonium, reprocessed uranium, and vitrified HLW. Currently, only the plutonium is recycled as MOX fuel to thermally-moderated reactors (LWRs). The thermal neutron spectrum limits the plutonium reuse to one or two cycles due to the ingrowth of higher plutonium and other TRU isotopes that do not fission under these conditions; current practice only performs one-cycle of MOX. MOX recycle increases the energy recovered by 50% or so as compared to the 1% uranium energy utilization with direct SNF disposal. In France the reprocessing output is coordinated with MOX plant input, to avoid accumulating significant inventories of plutonium. MOX is discussed further in Module 5.0, Section 5.1.10. The French experience indicates reprocessing with plutonium recycle is economical from the waste management perspective. The five-fold increase in the price of natural uranium over the past couple of years improves these economics. Reprocessed uranium is LEU but re-enrichment is needed for LWR use. Reprocessed uranium requires a slightly higher enrichment level than natural uranium enrichment to compensate for the reactivity dampening effects of uranium-236. The presence of uranium-236 also results im more neptunium-237 and plutonium-238 production during subsequent irradiation as reactor fuel; neptunium contributes to the long-term disposal toxicity of the subsequent SNF or HLW, while plutonium-238 contributes to the short-term heat load of any subsequently recovered plutonium Large-scale tests in a dedicated enrichment cascade have shown re-enrichment to

be practical. However, all currently generated reprocessed uranium is stored as the trioxide due to the current plentiful supply of natural uranium. The five-fold increase in the price of natural uranium over the past couple of years may change this perspective.

The original goal of reprocessing was to transition to a plutonium fuel cycle and fast breeder reactors. Such reactors operate in a fast neutron spectrum (neutron energies exceeding 1 MEV as compared to 1 ev or so for therma neutrons). In the fast spectrum, all plutonium and TRU isotopes fission to produce energy. Uranium-238 both fissions and absorbs a neutron to produce plutonium-239, which can also fission and generate more neutrons. Tests have demonstrated breeding - the production of more fuel materials (plutonium) over that consumed - due to the greater number of neutrons released by plutonium fission as compared to uranium fission. A breeder reactor/plutonium fuel cycle increases uranium energy utilization from the 1.5% or so with MOX recycle to 70-80% with plutonium fuel recycle. To put this in perspective, existing purified uranium at DOE sites, if used in a breeder reactor/plutonium fuel cycle, could supply all of the existing electrical needs of the U.S. for 1,000 years. At the present time, only limited efforts are investigating the breeder reactor/plutonium fuel cycle and no large scale efforts currently exist worldwide. This may change with the DOE GNEP Program (see Section 6.1.6).

Although not currently practiced to any significant degree, a thermal neutron spectrum breeding approach is also possible using thorium (primarily the 232 isotope, which is the most abundant in natural thorium) and uranium-233 as the fissile material. Two successive decays produce the uranium-233 from irradiated thorium-232 (i.e., from thorium-233). Essentially no TRU isotopes are produced and known world thorium supplies significantly exceed known uranium resources, but the breeding ratios are lower than with the uranium-plutonium route in the fast neutron spectrum. The approach has been demonstrated with experiments and test assemblies. Currently, India is investigating this cycle due to the country's significant indigenous resources of thorium.

Self-Check Questions 6-2

INSTRUCTIONS: Provide the correct response. Answers are located in the answer key section of the Trainee Guide.

1. What does reprocessing do?

2. Give examples of wet and dry reprocessing technologies.

3. Name a commercially operating reprocessing plant.

4. Identify some hazards of reprocessing.

5. What is a typical time period after SNF discharge for commercially reprocessing SNF today?

6. Discuss NRC regulations for reprocessing facilities.

7. Explain the breeder cycle.

You have completed this section.
Please check off your progress on the tracking form.
Go to the next section.

Learning Objective

When you finish this section, you will be able to:

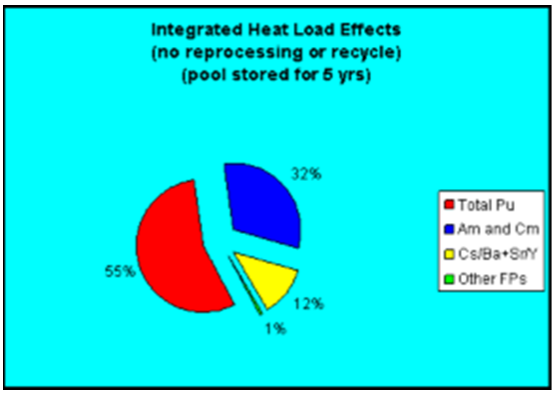
6.1.4 Identify high level waste (HLW).

6.1.5 Identify the repository for spent nuclear fuel and/or high level waste (Yucca Mountain).

HIGH LEVEL WASTE (HLW)

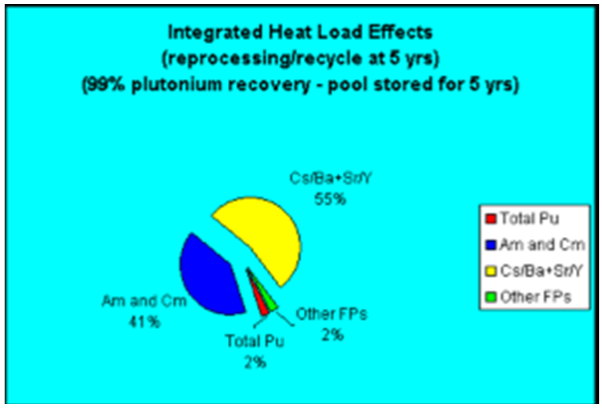
Reprocessing produces nuclear wastes which require great care in handling and management. HLW constitutes the main concern. HLW contains fission products and transuranic elements that emit alpha, beta and gamma radiation at high levels, as well as a lot of heat. Currently, the heat arises mainly from fission products, which have the shorter half-lives. However, plutonium and TRU produce the greatest overall integrated heat load and this affects disposal considerations. As shown in Figure 6-16, the recycle of plutonium significantly reduces subsequent heat loads.

Figure 6-16. HLW Integrated Repository Heat Loads



Potentially Big Impact:
- A repository could hold 4-10 times as much (HLW)
- Or heat load reduced
- Or smaller

Reprocessing and Recycle offers:
- 77% reduction in repository integrated heat load (Pu only)
- Additional 12% if Cs/Sr removed/recycled non-repository



Commercial reprocessing plants temporarily store liquid HLW in cooled multiple-walled stainless steel tanks surrounded by reinforced concrete (Figure 6-17 shows the U.S. carbon steel version). The tanks and piping are fabricated from stainless steels.

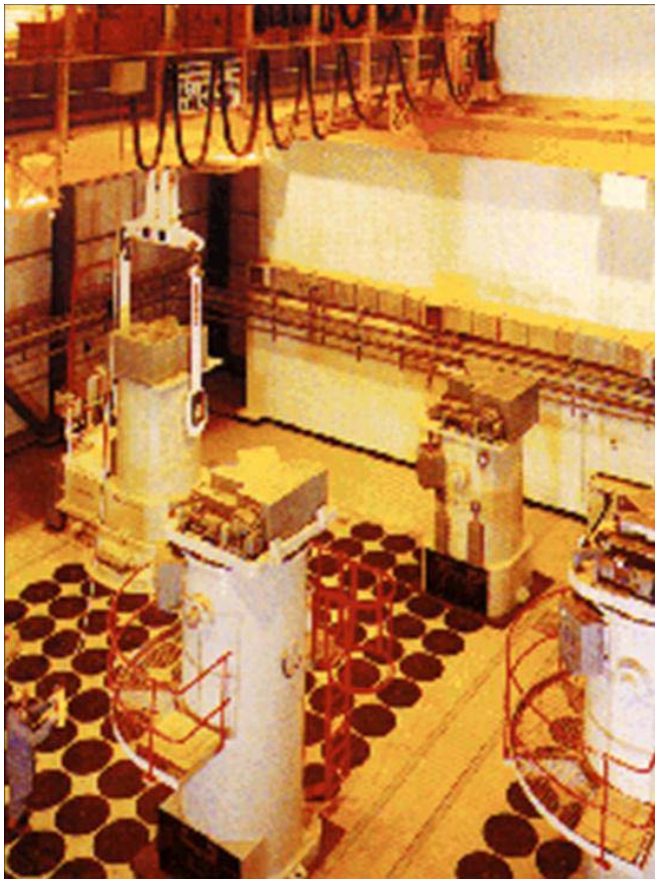
Figure 6-17. HLW Cans In Cell

Vitrification

The main method of solidifying liquid HLW is vitrification. The Australian synroc (synthetic rock) is a more sophisticated way to immobilize such waste, but this has not yet been commercially developed for civil wastes.

Vitrification converts the liquid HLW into a glass form by using high temperatures and borosilicate glass formers. Figure 6-18 displays vitrified HLW containers in a cooling cell, after vitrification. Multiple filled canisters are located below the shield-plugs in the floor. These are cooled by air.

Figure 6-18. UK Vitrified HLW Storage



Commercial vitrification plants calcine the wastes (evaporation to a dry powder) followed by incorporation in borosilicate glass in an induction melter. The molten glass is mixed with the dry wastes and poured into large stainless steel canisters, each holding 400 kg of HLW glass; the glass occupies about 150 liters of the 175 liter internal volume of the canister. A machine then welds a shield plug onto the opening. A year's HLW from reprocessing the SNF from a 1000 MWe reactor is contained in 5 tons of such glass, or about 12 canisters, each 1.3 meters high and 0.4 meters diameter. These are stored in air-cooled vertical silos (Figure 6-19). Thousands of these containers have been produced since the early 1990s.

Figure 6-19. Typical HLW Can

Processes such as these have been developed and tested in pilot plants since the 1960s. In the UK, several tons of HLW from reprocessed fuel were vitrified by 1966, but research was then set aside until there was enough HLW to give the issue a higher priority. High temperature leaching tests on this glass showed that it has remained insoluble even where some physical breakdown of the glass had occurred. Similar results have been obtained on French wastes vitrified between 1969 and 1972.

In the 1990s, two vitrification plants started operations in the U.S. using large, joule (resistance) heated melters; one at West Valley, NY to treat HLW from civil nuclear fuel reprocessed there 25 years earlier, and the other at the Savannah River Site, Aiken, SC, to vitrify a larger quantity of DOE waste. The West Valley melter has completed its mission and shut down. The DOE waste melter is located at Savannah River, and operations continue. The DOE canisters are larger than the French and English containers - about 10 ft high and holding about 625 liters of glass each. Canisters proposed for vitrified HLW from Hanford are 50% larger.

Vitrified wastes will be stored for some time before final disposal to allow heat and radioactivity to diminish. In general, the longer the material can be left before disposal, the fewer problems are involved and the less space is required in a repository. Depending on the actual disposal methods and sites adopted, there will be some 50 years between reactor and disposal.

All handling of such materials involves the use of protective shielding and procedures to ensure the safety of people involved. All transfer containers are shielded and loaded/unloaded automatically. Shield rings and plugs minimize personnel exposures during transfers. Very low radiation fields exist outside the shielded HLW canister or storage vault.

When separated high level wastes (or spent fuel assemblies) are moved from one place to another, robust shipping containers are used. These are designed to withstand all credible accident conditions without leakage or reduction in their radiation shielding effectiveness. Where such containers have been involved in serious accidents over the years, the robust container designs prevented any radioactivity hazards from occurring. The high standards of integrity designed into these containers also makes them difficult to breach with explosives and therefore unattractive as an object for sabotage attempts.

Transmutation

Transmutation is the process performed to render radioactive wastes either nonradioactive or significantly less radioactive so that radiotoxic and disposal concerns are substantially reduced or even eliminated. Transmutation uses high energy particles—neutrons or protons—to change the existing nuclei into another isotope or element that has a shorter half-life or reduced radiotoxicity concerns. The most developed transmutation approaches utilize high energy neutrons in a fast spectrum reactor; the first approach uses high energy neutrons to induce fission in heavy elements, while the second approach uses neutron absorption in lighter elements with subsequent radioactive decay. Many transmutation experiments are currently ongoing. Current results indicate a potential radiotoxicity reduction of two orders of magnitude or so as compared to the original SNF. While this is significant, it does not eliminate the requirements for geologic isolation of HLW in a repository.

Some fission products and TRUs are radioactive/hazardous for 10,000+ years and environmentally mobile. Transmutation would convert these into stable, nonradioactive or short-lived materials. The quantities of the isotopes would be reduced. Actinide transmutation constitutes the main focus. Actinides preferentially undergo fission in a fast neutron spectrum, as shown in Figure 6-20 for americium, a major isotope of concern. Thus, a net consumption occurs as compared to the formation of higher actinides. Experiments are underway to fully explore the spectrum of effects for the other TRU and quantify reductions. A secondary focus is on the long-lived fission products technetium, iodine, nickel, and zirconium. As illustrated in Figure 6-21, these routes utilize a neutron absorption/decay approach. The tertiary focus is on cesium and strontium.

Figure 6-20. Fission Transmutation

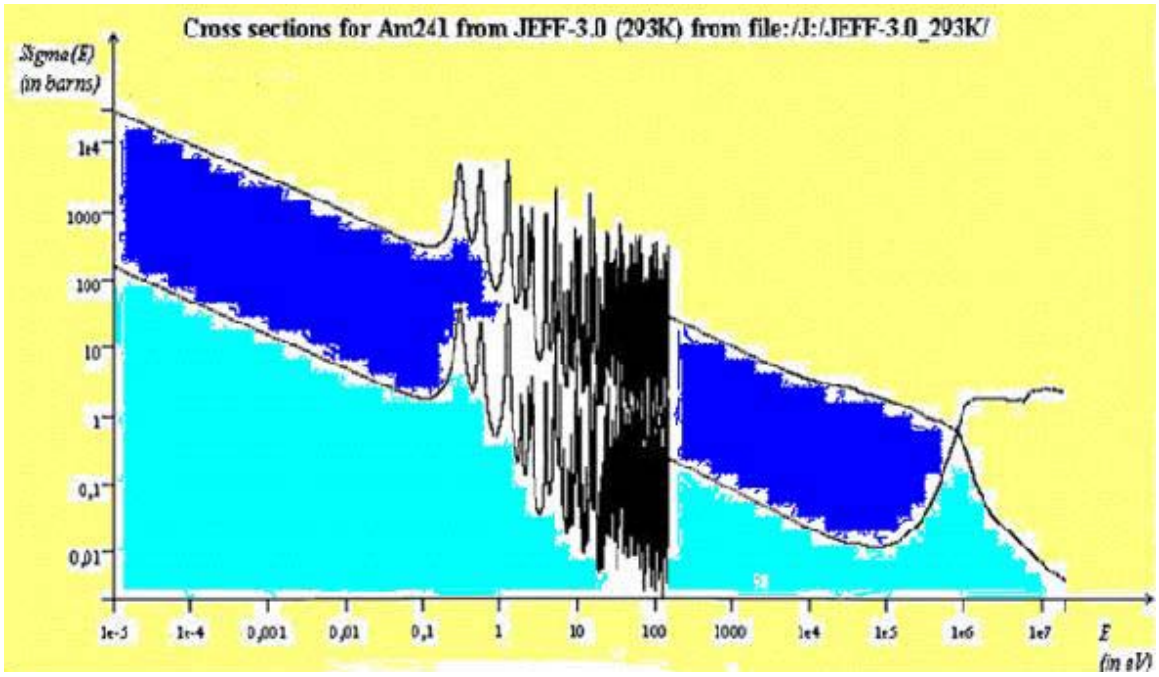
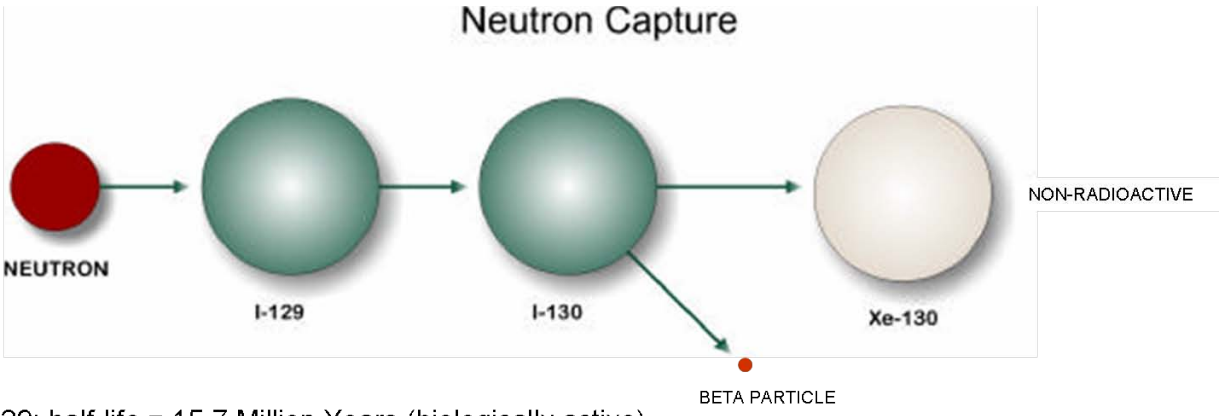


Figure 6-21. Transmutation of Iodine



I-29: half-life = 15.7 Million Years (biologically active)
I-30: half-life = 12.5 hours
Xe-130: non-radioactive and non-biologically active

REPOSITORY FOR HLW AND SNF

HLW and SNF contain sufficient quantities of radioactivity and long-lived isotopes that would result in potentially significant doses in near-surface disposal units. Consequently, HLW and

SNF require more rigorous disposal, normally referred to as geologic isolation in a rock formation. The proposed facility is referred to as a Federal Repository, or, more simply, a waste repository.

In the United States, a process has been followed to identify and develop such a disposal site for HLW and SNF. The Nuclear Waste Policy Act (NWPA) of 1982 directed DOE to develop a program for HLW and SNF disposal, including selection and evaluation of three sites, down selection to one site, development of the one site into the actual repository, transport of the HLW/SNF to the repository, waste emplacement, clsoure of the disposal site, and environmental monitoring requirements. The NWPA established a fee program for funding these activities, charging utilities a fee of 1 mill (0.1 cents) per KW-hr of nuclear generated electricity (ultimately, this fee was passed onto the utility customers and ratepayers). DOE organized the Office of Civilian Radioactive Waste Management (OCRWM - often identified as DOE-RW) for this program. The NWPA was amended in 1987 to focus on just one site—Yucca Mountain, Nevada. In 2002, DOE recommended Yucca Mountain as the repository for HLW and SNF, and this was approved for further evaluation and development of a license application.

The Acts identify the NRC as the regulatory agency for licensing the repository. However, for the proposed Yucca Mountain site, the NRC would regulate to offsite standards established by the EPA - these are 15 mrem/yr all pathways and 4 mrem/yr via the drinking water pathway. The NRC HLW regulation is specific to Yucca Mountain and is identified in 10 CFR Part 63. The Acts allow NRC to regulate to its own standards for subsequent repositories—these standards are 25 mrem/yr via all pathways (the NRC regulation is 10 CFR Part 60). DOE will be the license applicant to develop, operate, close, and seal the repository, and would use a contractor arrangement (GOCO - Government Owned, Contractor Operated).

DOE signed contracts with utilities to start accepting their SNF by January 1998. To date, DOE has spent approximately $8 billion on the program out of $25 billion collected in fees, and no SNF has been accepted by DOE. Utilities have filed numerous lawsuits. DOE has identified many estimates for acceptance of SNF at the repository—the current estimate is 2017.

DOE established a 10,000 year basis for evaluating potential doses from the repository. The 10,000 year time period arose from comparisons with the risks from natural uranium ore, in decay equilibrium (e.g., with radium-226). The uranium ore risk is constant due to the long half life of uranium. The risk from HLW and SNF decays below the uranium ore risk after several hundred years. This was rounded up to 1,000 years and a safety factor of 10 applied, resulting in the 10,000 year time period. Repository features, such as engineered barriers, vitrification, geology, and location, were considered to be additional features that would reduce the risk further below the uranium ore risk.

There have been many discussions, challenges, and lawsuits over the proposed repository at Yucca Mountain. A 2005 Federal Appeals Court decision found there was no clear regulatory basis for the 10,000 year time period, and that the EPA must consider the NAS (National Academy of Science) recommendation for a million year time period and provide a regulatory basis. EPA is working on the revised regulation and in 2006, issued a notice that included the same offsite radiation standard as before for the first 10,000 years and a 350 mrem/yr basis for the remainder of the one million year time period. The 350 mrem/yr value is based upon natural background.

Figure 6-22 is an aerial view of Yucca Mountain. Many studies—both analytical and experimental—are being performed. At Yucca Mountain, an exploratory tunnel has been drilled using a tunnel boring machine for actual thermal and hydraulic experiments with the rock material (Figures 6-23 and 6-24).

Figure 6-22. Aerial View of Yucca Mountain

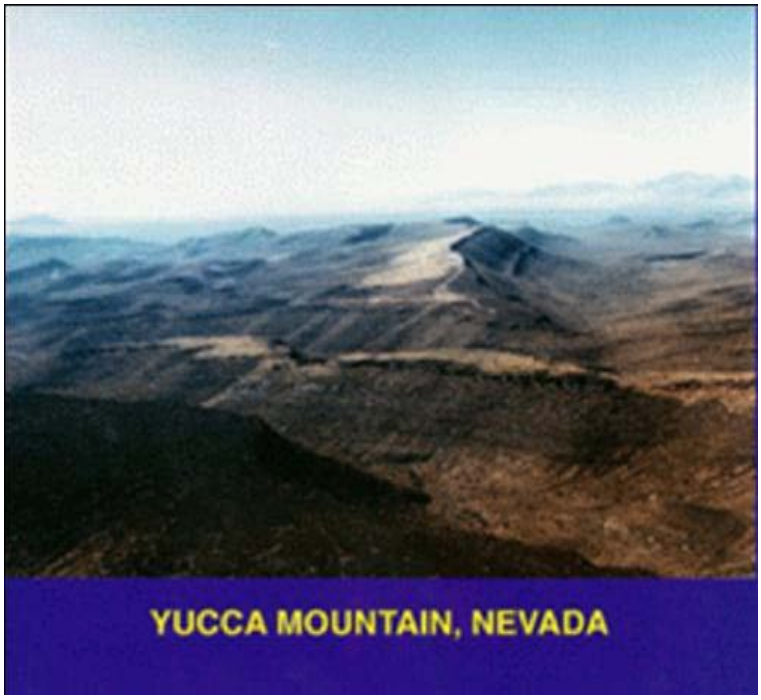


Figure 6-23. Exploratory Tunnel Entrance

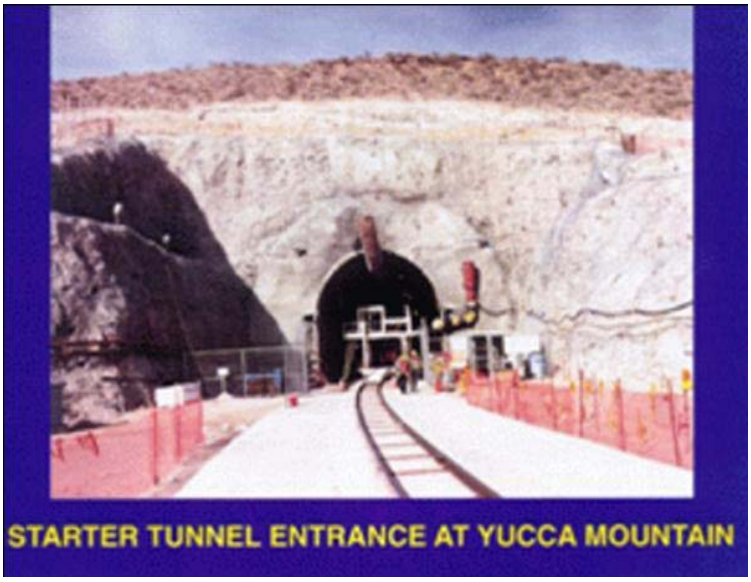
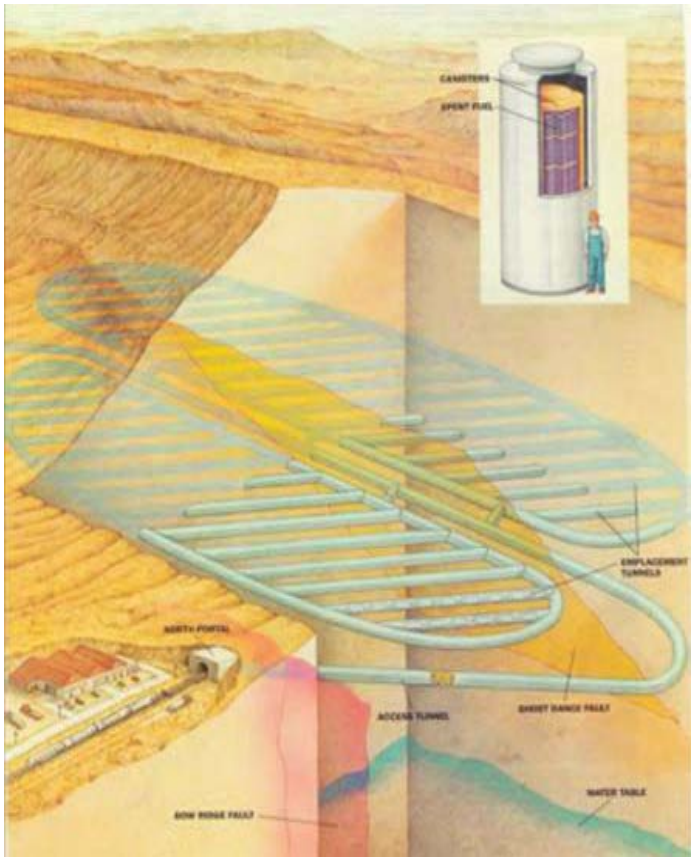


Figure 6-24. Tunnel Boring Machine (TBM)

DOE is evaluating designs and design approaches for Yucca Mountain. The main approach involves a single level repository underground with access and emplacement tunnels (Figure 6-25). Transport of waste packages containing HLW and SNF would be in a shielded transporter, but, after emplacement, the packages would be unshielded. DOE plans on a thermally hot design, nominally 57 KW/acre, that precludes the presence of liquid water near the waste packages for close to a thousand years. Conceptually, a hot design would eliminate corrosion and leaching for the time period water is not present. This means the adjacent rock temperatures are above 100 C for that period of time. Parametric analyses consider a super hot design (120 KW/acre - longer time period above 100 C) and a cold (25 KW/acre - the adjacent rock never goes above 100 C) design. The hot designs involve more temperature changes in the rock, with potential swelling and cracking, and, thus, require more experimental studies to determine rock behavior and to confirm the hypothesis of reduced corrosion and potential leaching. However, hot designs allow for a physically smaller repository or more total waste emplacement.

Figure 6-25. Schematic of Access and Emplacement Tunnels at Yucca Mountain



The Acts identify a statutory limit of 70,000 MTIHM for SNF emplacement in Yucca Mountain. From a technical basis, thermal loading and heat transfer parameters, and their assumptions, are the key parameters limiting the quantity of SNF that can be emplaced. DOE analyses indicate a physical limit of 100,000 to 120,000 MTIHM for Yucca Mountain, based upon a sealed repository and conductive heat transfer through the rock. The Electric Power Research Institute (EPRI) analysis shows perhaps a 300,000 to 500,000 MTIHM physical limit, based upon a multi-level design and convective cooling via the access tunnels. New reactors will increase the annual generation rate and inventories of SNF. Advanced LWR designs will likely produce similar quantities of SNF for the same power ratings. However, advanced gas reactors contain significant amounts of carbon-14 and may produce significantly larger volumes of SNF per unit power produced - perhaps an order of magnitude more. This may influence repository design and require some form of SNF conditioning or even reprocessing prior to disposal.

Estimates for repository conditions and performance vary considerably due to the many variables involved and possible assumptions. A brief overview is provided here to show key points - for more information, it is recommended the reader consult DOE-OCRWM and other related sources of information on the repository.

As previously mentioned, temperatures and heat loadings affect the repository and its capacity for HLW and SNF. Figure 6-26 displays the repository heat rate based upon several loading rate assumptions. The total heat loading increases as waste is emplaced. Significantly, even in the near term, the TRU isotopes generate a significant fraction of the total heat and their relative contribution is increasing. Estimates indicate the TRU isotopes contribute 80% of the total integrated heat load for the repository. The heat loading affects temperature. Figure 6-27 estimates temperatures for the direct disposal of SNF.

Temperatures remain relatively low with the air flow on during repository operations. However, termination of fan operation changes the heat transfer from convection to conduction and results in a rapid temperature increase and a 3,000 year time period with temperatures above boiling for the rock adjacent to the waste package. Figure 6-28 displays a repository effect from reprocessing; the majority of the TRU isotopes have been removed and the rock temperature remains above boiling for a much shorter time period (200 years for this example and assumptions). Figures 6-29, 6-30, and 6-31 show specific repository heat loads (per assembly or equivalent) by the major isotope groups for no reprocessing and reprocessing, respectively, and the heat load effects. Again, TRU isotopes are major contributors and their removal by reprocessing has a pronounced effect.

Figure 6-26. Repository Heat Rate

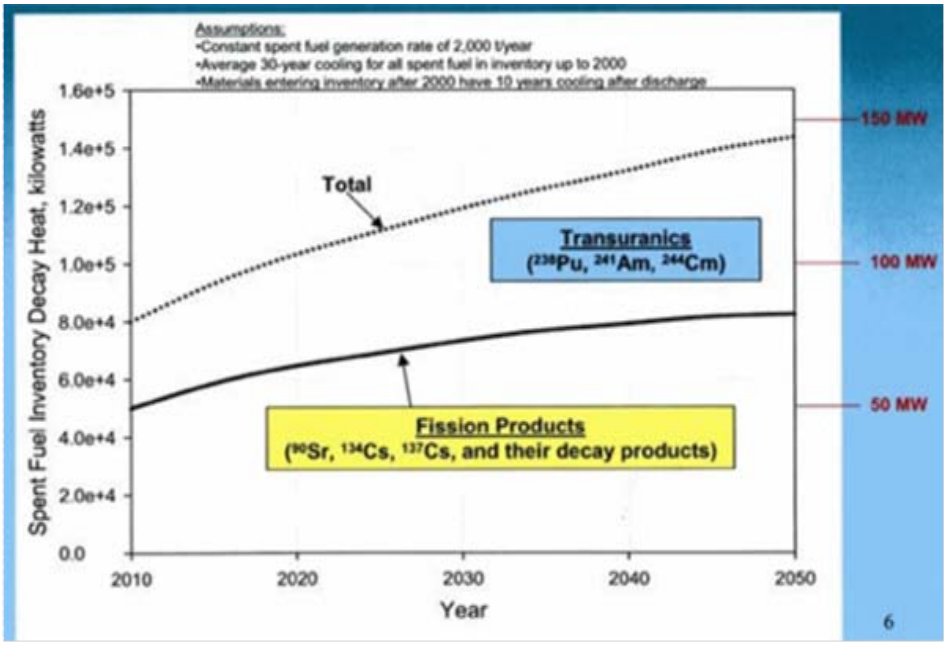


Figure 6-27. Temperature Estimates - Direct Disposal of SNF

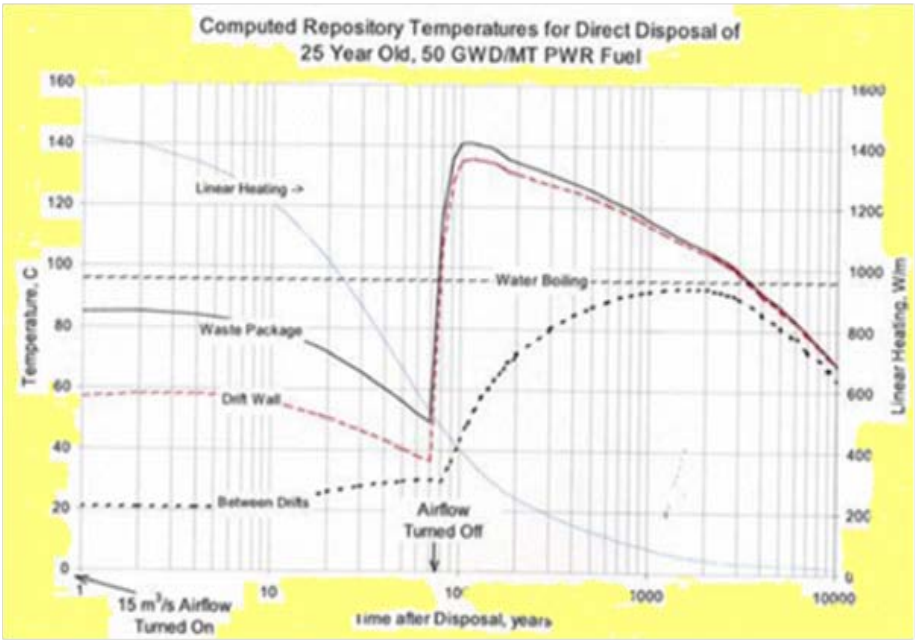


Figure 6-28. Temperature Estimates - GNEP SNF/HLW

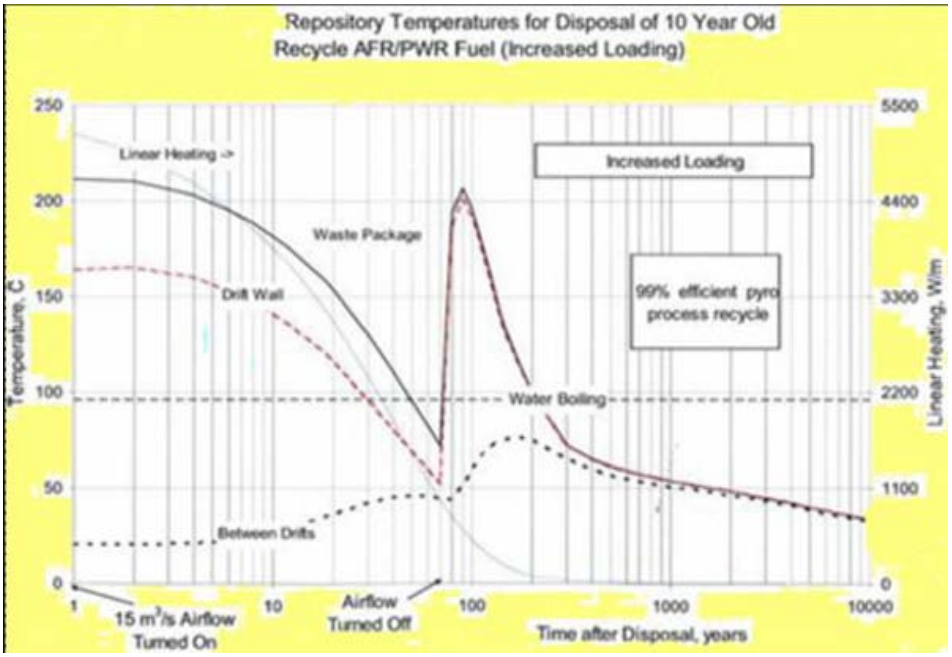
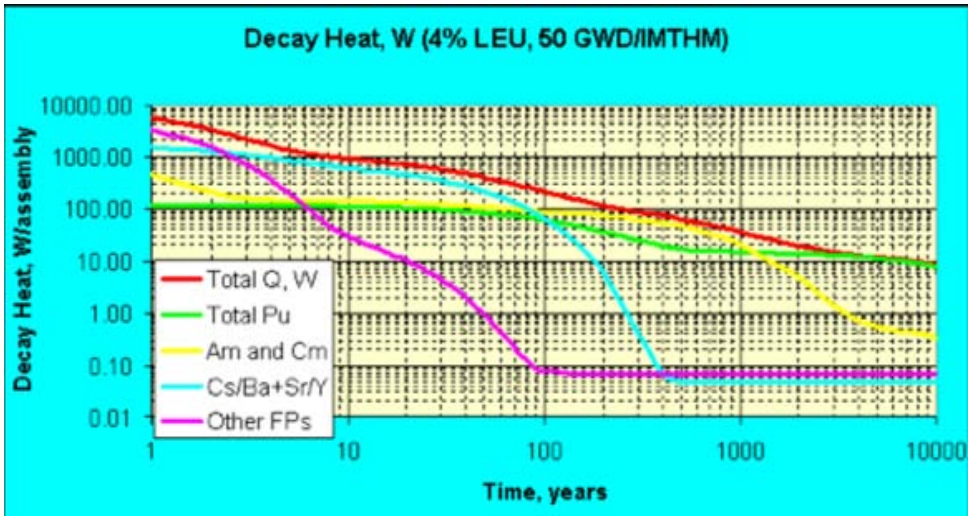
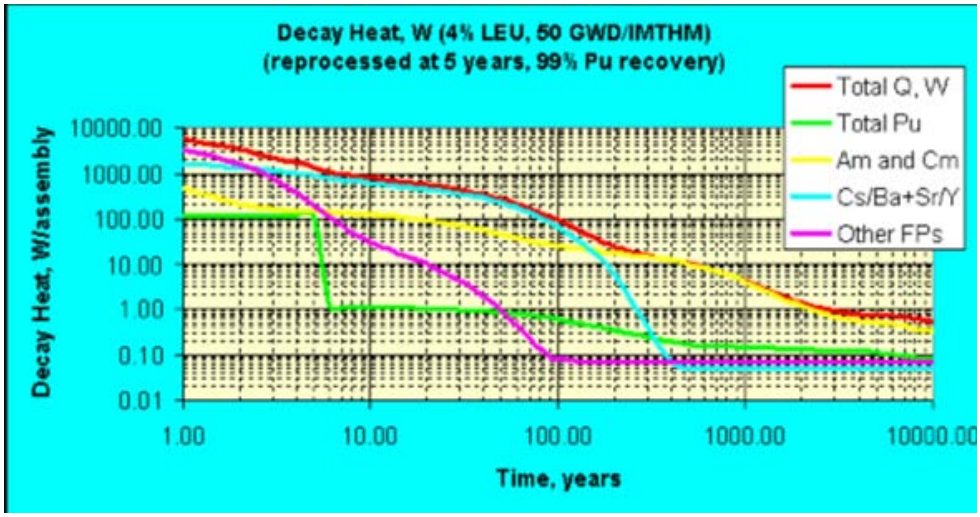


Figure 6-29. SNF Decay Heat - No Reprocessing



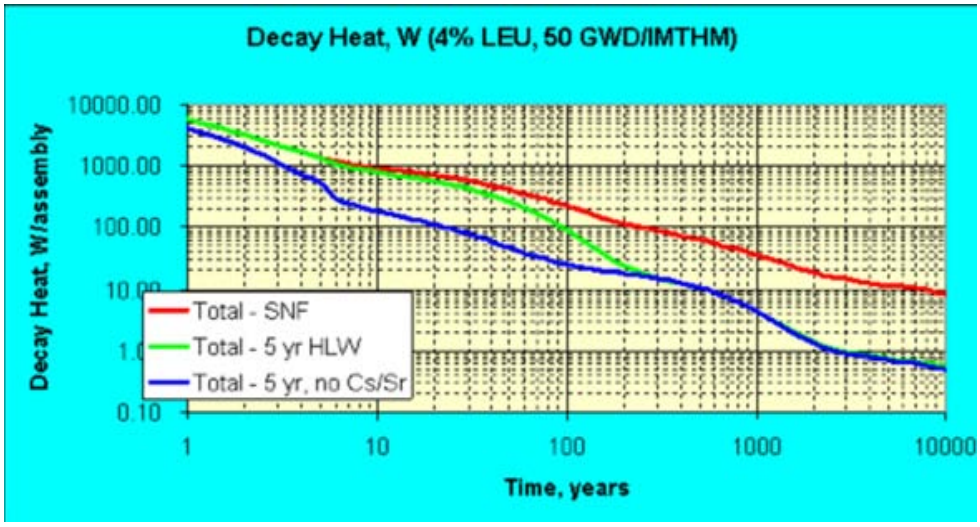
Note: 1 assembly–475 kg IHM

Figure 6-30. Reprocessing Effect on Major Isotopes



Note: 1 assembly–475 kg IHM

Figure 6-31. Reprocessing Effect on Heat Load



Note: 1 assembly–475 kg IHM

Pathway and performance analyses estimate doses to individuals. Again, modeling incorporates many parameters and assumptions. Figure 6-32 illustrates one example of dose estimates that presents typical findings for SNF disposal in the proposed repository. Doses remain very low for the initial 10,000 year period but start increasing rapidly due to container failure and the initiation of leaching. Doses continue to increase beyond the 10,000 year period, and many parametric analyses show an actual peak in the 100,000 to 200,000 year range, sometimes with a dose exceeding 100 mrem/yr (all pathways). Significantly, the key isotopes producing these dose estimates are nonintuitive - technetium-99, iodine-129, and neptunium-237. Neptunium-237 is the dominant contributor to the dose and dose peak value.

Figure 6-33 provides estimates of repository capacity for SNF. At the present time, the current inventory of SNF is very close to the repository's statutory limit of 70,000 MTIHM (a small amount of this limit is consumed by DOE SNF); the statutory limit will likely be exceeded in the next few years. DOE estimates the physical capacity of the repository at around 100,000 MTIHM, based on a single level design. Thus, for practical purposes, the proposed repository is essentially full, based upon current planning and design parameters. Additional SNF disposal capacity can only be obtained by making the proposed repository larger (physically or by design assumption changes), having more repositories, or by reprocessing.

Figure 6-32. Dose Estimates

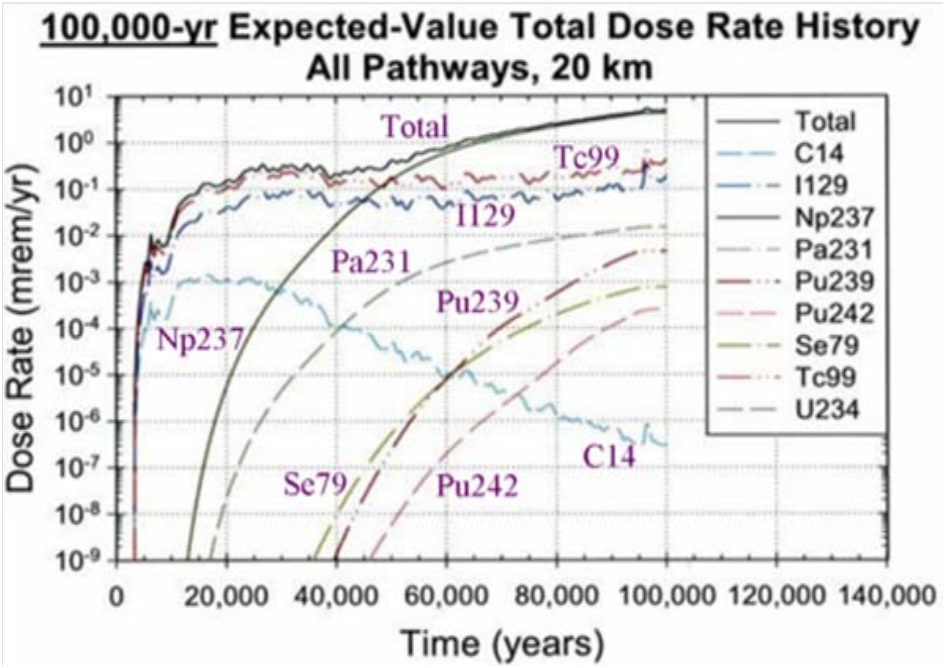
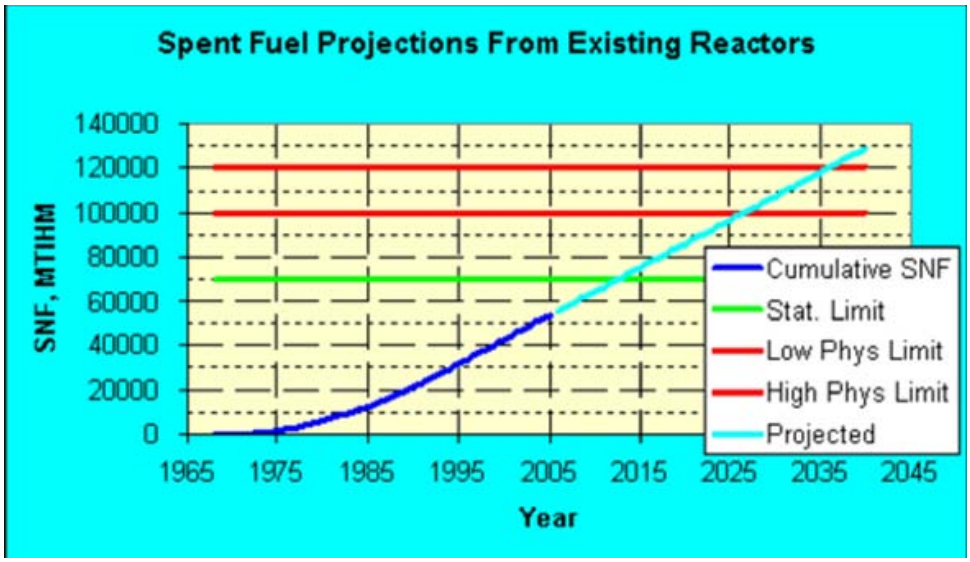


Figure 6-33. Repository Space



Self-Check Questions 6-3

INSTRUCTIONS: Complete the following questions. Answers are located in the Trainee Guide.

Complete the following questions. Answers are located in the Trainee Guide.

1. Identify the NRC regulations for HLW and SNF disposal in a repository.

2. These two groups of isotopes constitute the main radioactivity in HLW and SNF.

3. State the three isotopes responsible for the main dose contributions from SNF disposal in the proposed repository.

4. Name the two isotopes with the highest near-term impact on heat loading.

5. Name the isotopes or isotope group with the largest integrated heat impact upon SNF disposal in a repository.

6. Identify the two main transmutation approaches currently being considered.

7. What are the two types of melters currently used for HLW vitrification?

8. Where is the proposed Federal repository for HLW and SNF disposal located?

9. What is the principal parameter determing SNF quantity limits for emplacement in a repository?

10. State the statutory limit for SNF in the repository.

You have completed this section.
Please check off your progress on the tracking form.
Go to the next section.

MODULE 6.0: BACK-END OF THE FUEL CYCLE: SPENT NUCLEAR FUEL AND IRRADIATED MATERIALS

Learning Objective

When you finish this section, you will be able to:

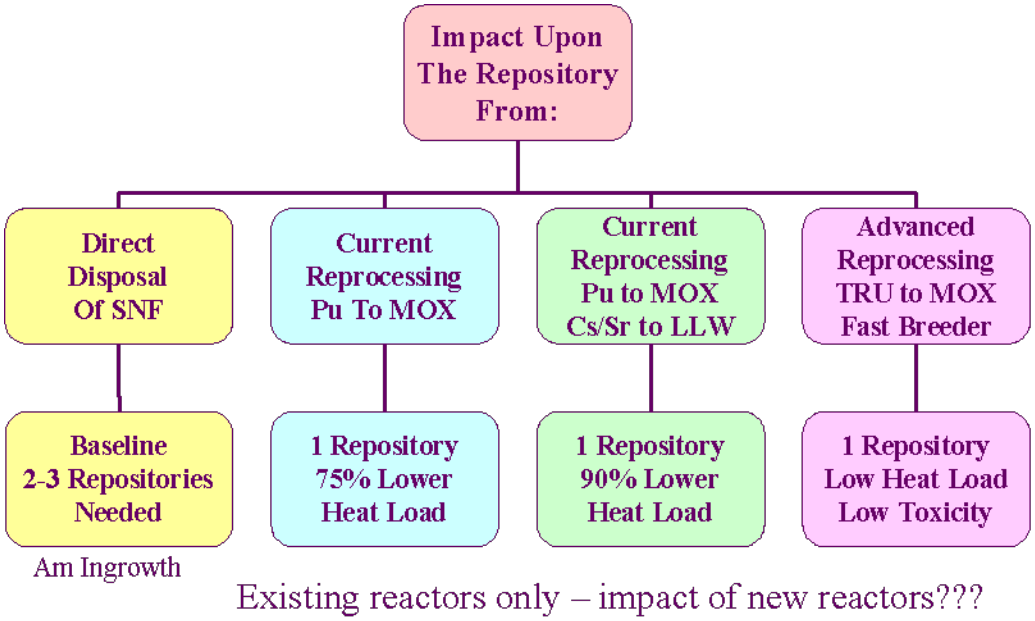
6.1.6 Identify ongoing programs and NRC involvement with SNF.

ONGOING PROGRAMS

The problem of managing spent nuclear fuel is one faced by all countries that operate nuclear power facilities. The DOE, IAEA, and others have recognized that spent fuel is accumulating in many geographic regions, in a decentralized fashion and according to varying standards, and represents a potential concern for safety, the environment, and global security. There appears to be a concensus forming to institute some form of international program or programs that will address SNF issues and expand the use of nuclear power.

In the U.S., DOE has initiated the GNEP program to address these concerns. DOE recognizes there are repository limitations and that fuel cycle and reactor decisions influence the repository significantly. Figure 6-34 shows some of the repository impacts from SNF and fuel cycle decisions, such as reprocessing and recycle. GNEP has many goals, including reducing the quantities and toxicities of materials entering a repository. Reprocessing of SNF and recycle into fast spectrum reactors for energy production and transmutation are key parts of the GNEP program. GNEP will involve commercial facilities that will require licensing by the NRC.

Figure 6-34. Repository Impacts



Wastes from weapons programs will continue to influence the scene in countries like the U.S. and Russia for many decades, no matter how rapidly commercial nuclear power expands. These defense related wastes consume repository space and may use similar processing facilities (e.g., vitrification facilities).

Costs

Many studies have been performed on U.S. fuel cycle costs and the contribution from the back-end of the fuel cycle. These studies have consistently implied that, while absolute monetary costs are in the tens of billions of dollars and may even approach $100 billion (current U.S. reactors only), the total is only a few percent of the cost of generating electricity from nuclear power. It is likely the economic differences between reprocessing and direct disposal of SNF are small or marginal from a waste management perspective. However, from resource, toxicity, and energy perspectives, the differences are significant and greatly favor reprocessing. Currently, DOE assesses a fee of 1 mil per KW-hr of nuclear generated electricity fee for projected SNF costs associated with the repository. Some $25 billion has been paid into this nuclear waste fund, while approximately $8 billion has been spent on repository exploration, testing, and development activities, but no SNF has been removed from reactor sites. For comparison, $8 billion is approximately equivalent to the current commercial charges for reprocessing 8,000 MTIHM of SNF.

NRC INVOLVEMENT

There are currently 104 (2006) licensed power reactors in the U.S. of which 103 are operating and producing electricity. Utilities have publicly stated plans to build an additional 26 new reactors over the next decade or so. As part of its reactor licensing activities, the NRC has to make a determination that a disposition pathway exists for any SNF that would be generated by the proposed new reactor—this is termed the "waste confidence rule." This determination would be simplified if the HLW and SNF approaches, and the repository were better defined and facility progress accomplished.

It's time to schedule a progress meeting with your administrator. Review the progress meeting form on the next page. In Part III, as a Regulator, write your specific questions to discuss with the administrator.

PROGRESS REVIEW MEETING FORM

Date Scheduled: _____Location:_____

I. The following suggested items should be discussed with the administrator as to how they pertain to your current position:

II. Use the space below to take notes during your meeting.

III. As a Regulator:

Use the space below to write your specific questions.

IV. Further assignments? If yes, please note and complete. If no, initial completion of progress meeting on tracking form.

Ensure that you and your administrator have dated and initialed your progress on your tracking form for this module.
Go to the module summary.

MODULE 6.0: BACK-END OF THE FUEL CYCLE: SPENT NUCLEAR FUEL AND IRRADIATED MATERIALS

MODULE SUMMARY

Key Points:

- The "back-end" of the fuel cycle, i.e., post-reactor use of the (now) irradiated fuel, addresses the management and disposal of spent nuclear fuel (SNF) and related radioactive wastes. The handling of SNF and its related radioactive wastes is a problematic, complex and controversial aspect of the nuclear fuel cycle today.
- SNF is currently stored at nuclear power plants, primarily in spent fuel pools but also in dry storage containers. The current inventory of SNF just exceeds 50,000 MTIHM, and is accumulating at 2,000-2,500 MTIHM/yr.
- SNF has considerable value. It contains stable, non-radioactive fission products, such as the platinum group metals and xenon. The energy equivalent content of the current SNF inventory would supply the entire electrical needs of the U.S. for around 100 years, at current levels, if recycled into a fast spectrum reactor. If SNF were used in fast spectrum breeder reactors with the existing, U.S. inventory of depleted uranium, the energy content would be equivalent to 1,000 years of the current electricity usage. Some countries are using reprocessing for recovering this energy value.
- Yucca Mountain, Nevada is the proposed site for the disposal of SNF in the U.S. Many studies and evaluations remain to be performed on the repository. However, it is essentially at or near projected capacity with the existing SNF from existing nuclear power plants, and, if this route is followed, more repositories would likely be needed. Reprocessing offers the likelihood of reduced repository capacity needs and reduced future doses from a repository.
- DOE has initiated a program GNEP that includes reprocessing and recycle to fast spectrum reactors as the means to increase energy utilization, reduce repository heat load and quantity needs, and reduce future doses from a repository.

Congratulations! You are ready to go to the next assigned module.
