Alternatives For Managing Wastes From Reactors And Post-Fission Operations In The LWR Fuel Cycle

MASTER

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Volume 1:

Summary

Alternatives For The Back End Of The LWR Fuel Cycle

Types And Properties Of LWR Fuel Cycle Wastes

Projections Of Waste Quantities

Selected Glossary

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United States
Energy Research & Development
Administration

Volume 1
(1 of 5 Volumes)

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Report Coordinated By
Battelle, Pacific Northwest Laboratories
At The Request Of The
Division Of Nuclear Fuel Cycle And Production

FOREWORD

This report describes technical alternatives for managing wastes from the back end of the commercial LWR fuel cycle. It describes waste types, the alternative technologies that can be used to manage the wastes, and the state of availability of those technologies.

The report does not select preferred waste management technologies or make comparative assessments. Its purpose is to provide a comprehensive digest of technical information expected to serve as a basis for future decisions, assessments, and environmental impact statements. The technology is still evolving, and this report will be subjected to public review and comment. Its contents therefore serve as a baseline subject to future supplementation and updating.

There are four major waste management functions: treatment, interim storage, transportation, and final storage or disposal. This report is structured in five volumes that reflect these functions. Volume 1 summarizes the report, describes alternatives for the back end of the LWR cycle, and provides background information on waste types and quantities. Volume 2 describes alternatives for waste treatment. Volume 3 discusses interim storage and transportation. Alternatives for final storage and disposal are described in Volume 4. The Appendices, Volume 5, provide supplementary information, with emphasis on characteristics of geologic formations that might be used for final storage or disposal.

Preparation of this report was started in August of 1975, in response to a directive from ERDA's Nuclear Fuel Cycle and Production Division. The report was authored by personnel of ERDA's contractors, of industry, and of universities; coordination was provided by Battelle's Pacific Northwest Laboratories. The report was completed in March of 1976. The cutoff date for characterizing the status of the technologies was September 1, 1975.

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INTRODUCTION
AND EXECUTIVE SUMMARY

INTRODUCTION AND EXECUTIVE SUMMARY

This document describes technologies for managing radioactive wastes from commercial nuclear power. It is directed specifically at wastes associated with what is known as "the back end of the LWR fuel cycle," i.e., activities related to handling of spent fuel after it is removed from light-water-cooled reactors (LWRs). The LWRs are currently the predominant reactor type in the commercial U. S. nuclear power industry.

The document characterizes the technologies and classifies their state of availability. It is not a decision or program document. It is strictly a baseline or reference document, technical in nature, and designed to serve as a sourcebook of information on the technologies of radioactive waste management. It is a bridge between past and future activities to develop technology for managing radioactive wastes; i.e., it describes the current status of technology and provides a basis for future action. It is assumed that the technologies to be adopted must be operable within the applicable radiation safety regulations.

ORIGINS AND FUNCTIONS OF THE DOCUMENT

This document is an outgrowth of a need for technical alternatives in preparation of an environmental statement and a request from the Joint Committee on Atomic Energy for a comprehensive report on options for managing radioactive wastes. It is also a step in ERDA strategy for fulfilling its responsibilities concerning development of commercial radioactive waste management technology. The document is responsive to the JCAE request for a discussion of options; it simultaneously provides baseline information and data for formulation of ERDA program plans and preparation of environmental impact statements. Other documents will address R&D plans; still others will address environmental impacts.

The basic purpose of this document is to provide government, industry, and the public a convenient, comprehensive and non-evaluative description of the technologies to be considered in actions such as preparation of environmental impact statements or selection of ERDA programs concerned with managing wastes from the back end of the commercial LWR fuel cycle. A document of this type is needed because there are in fact numerous alternative technologies for each aspect of radioactive waste management. This compilation of information helps assure that all alternatives are considered, and that all participants in actions to make selections have a common information base. To preserve its role as a source document, this report makes no recommendations, makes no comparative assessments

and program selections are future actions to be derived from the contents of this document and public discussion of the alternatives.

BACKGROUND FOR THE TECHNOLOGY ADDRESSED

Spent LWR fuel contains fissionable uranium and plutonium in concentrations that are potentially of commercial interest. To make use of these materials, the fuel would be processed to recover the uranium and plutonium; these materials would then be fabricated into fuel elements containing mixed oxides of uranium and plutonium (MOX fuel elements) which would be recycled to reactors.

If uranium and plutonium are recovered and recycled, the back end of the LWR fuel cycle is said to be closed. One alternative to this mode of operation for the back end of the fuel cycle is to regard the spent fuel as material to be stored indefinitely or discarded; this is known as the "throwaway" fuel cycle. Other alternatives involve recycle of one or the other of the fuel materials, e.g., recycle of uranium and storage of plutonium.

Radioactive materials in the back end of the fuel cycle become wastes when they are to be discarded. In their initial form, the wastes occur as a variety of solid, liquid, and gaseous materials; these are called "primary" wastes.

The primary wastes may undergo treatment prior to final disposition. For example, liquids can be converted to solids, and combustible materials can be burned in order to reduce their volume. In general, such operations are known as "treatment technologies"; there are numerous alternative specific concepts for each type of treatment.

Treatment of primary wastes produces what are known as "secondary" wastes which consist of the treated primary wastes plus any other wastes that result from the treatment process itself. Combustion, for example, generates off-gases that may be filtered prior to discharge; the spent filters, in addition to the residue from combustion of the primary wastes, become part of the secondary wastes.

In general, the treatment operations take a broad spectrum of primary waste materials and convert them to a few secondary waste types in forms suitable for transportation, handling, and final disposition. Quantities and characteristics of the secondary wastes will depend on which treatment technologies are used, which fuel cycle mode (e.g., throwaway or closed) is used, and what use is made of logistic factors such as holdup times prior to treatment.

A single, unique characterization of commercial radioactive waste management is therefore not possible. Specific fuel cycle conditions and waste management technologies must be defined from among the alternatives. This document identifies and characterizes the waste management technology alternatives; these can be regarded as alternative "building blocks" from which complete waste management systems can be assembled.

In addition to waste treatment there are three other basic waste management functions: interim storage, transportation, and final isolation or disposal. Alternatives for each are discussed in this document. The terminology is used as follows:

- Storage is "interim" when the waste is being held with anticipation of subsequent waste management action such as treatment or transportation. An example of interim storage is tank storage of liquid wastes prior to solidification.
- Transportation is movement of materials from one site to another.
- "Final isolation or disposal" is the terminology used for the alternatives for final disposition of the wastes. "Disposal" refers specifically to alternatives that offer little or no chance for reversibility. "Final isolation" refers to alternatives that isolate the wastes from mankind and the biologic environment by placing them in stable geologic media.

Some of the isolation alternatives involve disposal concepts; i.e., the wastes would be put into geologic isolation using emplacement techniques not intended to permit reversibility. Other isolation alternatives involve "final storage," which isolates the wastes from the biologic environment by using emplacement techniques that do offer opportunity for reversibility, i.e., retrieval.

Two types of geologic isolation storage can be distinguished: "provisional geologic storage," which uses emplacement techniques and monitoring operations that permit retrieval with methods and rates essentially the same as those used for emplacement, and "permanent geologic storage" from which the wastes could be retrieved using excavation and mining techniques. Repositories operated for provisional storage could be converted to permanent storage by backfilling and sealing.

This document discusses four basic types of media that can be used for geologic isolation by storage or disposal: shallow continental geologic formations, deep continental geologic formations, the sea floor, and ice sheets. It also describes two other irreversible disposal alternatives: ejection to space and transmutation, which involves bombarding the waste materials with nuclear particles or radiation in order to convert them to other isotopes. The latter two alternatives can be used only for selected waste constituents which would have to be separated from the rest of the materials in the waste that contains them.

In summary, alternatives for the back end of the LWR fuel cycle consist of options for the fuel cycle mode and options for technologies used for treatment, interim storage, transportation, and final isolation or disposal of wastes. This document describes the alternative waste management technologies and their current status. It is a technical basis for future programs, public discussion, and environmental impact statements.

SCOPE OF THE DOCUMENT

This document describes alternative technologies for all steps of management of all wastes for the back end of the LWR fuel cycle, including wastes from the reactors and wastes produced by decommissioning fuel cycle and waste management facilities. Technologies being developed or used in other nations are included.

The report also includes description of LWR fuel cycle alternatives, description of wastes, and projection of waste quantities. In practice, waste quantities and types, and the waste properties, will depend on many things, including the operating history of the reactors; the length of time the spent fuel is held in storage; the mode of operation of the back end of the fuel cycle (e.g., throwaway or closed); the technology, if used, for processing the spent fuel; the technology used for waste treatment and the length of time wastes are held in interim storage. The descriptions of wastes and projection of waste quantities that are included here are estimated on the basis of known characteristics of the technologies and anticipated operating practice in the industry.

For purposes of illustrating the effect of technology and fuel cycle mode on waste quantities and properties, the report also includes overview characterizations of three alternative fuel cycle modes: operating without spent fuel processing, recycle of uranium coupled with storage of plutonium, and recycle of both uranium and plutonium. In all three cases, use of current waste management technology is assumed.

The descriptions of waste management technologies given in this document are, for the expert professional, highly condensed. Readers interested in details may consult the references or authors. References have been used extensively but without attempt to provide exhaustive compilation of the waste management literature. The references were selected to provide access to additional literature and information.

STATUS OF THE TECHNOLOGY

A major purpose of this report was to characterize waste management technology in terms of its state of readiness for commercial use. Another purpose was to determine if there are any areas where needed waste management technology is missing, i.e., not now available or under development. The cutoff date for characterizing the status of technologies was September 1, 1975.

The status of technologies is characterized and classified as commercialized, available, or under development. Commercialized technology is in routine use with radioactive materials for industry purposes at economic scale. Technology is "available" if its status is such that design and construction of a full scale commercial installation can be initiated; design verification tests might be required. The status of technologies under development can range from early-on concept testing in the laboratory to nearly available.

The information contained in this document shows that all technologies needed to manage radioactive wastes from the back end of the commercial LWR fuel cycle are commercialized, available, or under development; there are no gaps. Management of radioactive wastes from all options for the back end of the LWR fuel cycle can be accomplished using commercialized or available technology as defined above. Technologies for managing wastes from LWR reactors are fully commercialized. Technologies for treatment, interim storage, and transportation of wastes from fuel cycle operations such as fuel processing are commercialized, or ready for commercial-scale design and proof-testing, or can readily be implemented by adaptation of commercial practice for management of nonradioactive wastes.

The available technologies for final disposition of wastes are burial grounds and provisional storage in deep continental geologic formations. Stable geologies expected to be suitable for deep geologic isolation are known, technologies for site exploration and site selection are available, and design principles for waste repositories are known. Repository designs and qualification procedures will be specific to the site being considered; they can be developed as necessary when candidate sites are identified.

The technologies under development provide alternatives to existing technology. Implementation of the developing technologies would be expected to provide benefits such as reduced environmental impacts, simplified procedures, and reduced costs in comparison with existing technology. Comparative assessments of existing and developing technologies can be made as information from the development work becomes sufficient for such evaluations.

DOCUMENT STRUCTURE

This document is structured to reflect the four basic waste management technologies of treatment, interim storage, transportation, and final isolation or disposal. Volume 1 includes, in addition to the Introduction (Executive Summary) and a Technical Summary, background information on the LWR fuel cycle alternatives, descriptions of waste types, and projections of waste quantities. Volume 1 also contains the overview characterizations of alternative LWR fuel cycle modes.

Volume 2 describes alternative waste treatment technologies. Volume 3 describes alternatives for interim storage and transportation. Volume 4 describes final isolation and disposal alternatives. The Appendices, Volume 5, provide supporting information, with emphasis on continental and sea floor geologies.

TECHNICAL SUMMARY

This is a technically oriented summary of this document, which describes alternatives for managing radioactive wastes from reactors, spent fuel storage basins, fuel reprocessing plants, and mixed-oxide fuel fabrication plants in the fuel cycle for light-water-cooled commercial reactors (LWR's). In commonly used terminology, the document describes waste management alternatives for the back end of the commercial LWR fuel cycle.

This document characterizes the technologies and classifies their status; it is not a decision or assessment document. It is a digest of technical information on waste management alternatives which is expected to serve as a basis for future decisions, assessments, and environmental impact statements. It is assumed that technologies to be adopted must be operable within the applicable radiation safety regulations. Because of its focus on the back end of the commercial LWR fuel cycle and its role as a reference document, the document does not:

- . address pre-fission waste management issues
- . address management of noncommercial (e.g., Defense) wastes
- . address environmental impact issues
- . select a preferred commercial waste management system
- . estimate and compare costs
- . estimate and compare safety and risks.

The scope of this document does include:

- . identification and characterization of wastes from the back end of the commercial LWR fuel cycle
- . projection of waste quantities
- . identification and description of alternative technologies for treatment, interim storage, and transportation of wastes
- . identification and description of alternatives for final storage and disposal
- . overview characterization of three alternative fuel cycle and waste management systems:
 - operation without fuel reprocessing ("throwaway" fuel cycle)
 - recycle of uranium, storage of plutonium, and use of available waste management technology
 - recycle of uranium and plutonium; use of available waste management technology.

These characterizations illustrate comparison of alternative LWR fuel cycle modes operated with the same waste management technology. Technologies which are not now ready for application but which show promise of economic, environmental impact, or operational benefits are

candidates for research and development and detailed assessment of their effects on fuel cycle and waste management characteristics and impacts.

The alternative waste management technologies are discussed in terms of four basic waste management activities:

- . Treatment, which converts the wastes from their as-generated form to forms needed for handling, transportation, and final disposition of the wastes. As-generated wastes are termed "primary wastes"; wastes that emerge from treatment operations are "secondary wastes."
- . Interim storage, which provides retention of primary and secondary wastes while they are in man's environment and subject to subsequent management activities such as treatment or transportation.
- . Transportation, which is applied to movement of waste materials between sites.
- . Final isolation or disposal, which provides final disposition of wastes. The basic alternatives are isolation in geologic media or elimination of the wastes from existence on earth.

Anticipated waste types, quantities, and characteristics for the commercial LWR fuel cycle are summarized in Figure 1. Actual waste quantities will depend on the specific technologies used in fuel cycle operations and waste management. Values given in Figure 1 are representative of current practice. Waste types and radioactivity characteristics shown in Figure 1 are expected to be generally representative of primary wastes generated by the commercial LWR fuel cycle.

Alternatives for the fuel cycle and waste management can be summarized as follows:

FUEL CYCLE OPERATIONS

Characteristics and quantities of wastes produced in the back end of the LWR fuel cycle will depend on technologies and logistics used for fuel cycle and waste management operations. Overview characterizations were developed for three basic alternatives, each of which was assumed to use available waste management technology:

- . No recycle of uranium or plutonium.
- . Uranium recycle only (plutonium is stored).
- . Recycle of uranium and plutonium.

Waste management requirements for these alternatives differ in detail. The two recycle options are, within the framework of assumptions used, similar with respect to impacts on waste management: both require fuel reprocessing and its comparatively strong impact on waste quantities and characteristics. Recycle of plutonium in addition to uranium involves

*NOT DISCUSSED IN THIS DOCUMENT

FIGURE 1. Process Operations and Wastes in the LWR Fuel Cycle

addition of mixed-oxide fuel fabrication plants to the fuel cycle; such operations have minor effect on waste quantities. The "throwaway" fuel cycle (no recycle of plutonium or uranium) minimizes waste quantities but maximizes inventories of long-lived alpha particle-emitting nuclides in the wastes. Detailed comparative assessments of these alternatives have not been made and are not within scope of this document.

IRRADIATED FUEL

The technology is available and in use for interim storage and transportation of irradiated LWR fuel. If the irradiated fuel becomes a waste, i.e., if the uranium and plutonium in spent fuel are not returned to the fuel cycle, then the fuel assemblies can be stored or managed as a waste, e.g., by placing them in containers and filling the void space with metal. Such technology is available but has not been used.

CHOP-LEACH FUEL BUNDLE RESIDUES

The chop-leach fuel bundle residues are solid wastes comprising short lengths of fuel cladding, fuel bundle support rods, poison rods, massive end fittings, fuel support grids, assorted springs, and spacer elements. Between 0.05 and 0.5% of the original fuel material remains with these wastes as an insoluble residue. The available technology is to store these wastes without treatment. The small amount of chop-leach fuel bundle residue wastes generated in the U.S. to date, at the Nuclear Fuel Services Plant, was packaged in steel containers and placed in shallow geologic isolation (burial grounds).

Advanced technology is being developed to decontaminate and/or consolidate these wastes by mechanical compaction, by melting, or by chemical reaction.

HIGH-LEVEL LIQUID WASTE

High-level liquid waste (HLLW) is generated in the first solvent extraction cycle of nuclear fuel reprocessing facilities. Because HLLW can contain over 99.9% of the nonvolatile fission products and more than 50% of the actinide activity in the wastes from the back end of the fuel cycle, significant amounts of research, development, and engineering resources have been applied to this waste.

No LWR fuel cycle HLLW has yet been generated in the United States except for a small amount generated at the Nuclear Fuel Services plant at West Valley, New York, during the years 1966-1972. That waste is now in storage. Present regulations require that when more commercial HLLW is generated in the U.S. it must be converted to a stable, dry solid within five years after fuel reprocessing.

Technology exists to convert the HLLW to a calcine powder or a glass. ERDA wastes, which differ chemically from LWR fuel cycle wastes and have lower levels of radioactivity, have been calcined routinely at the Idaho National Engineering Laboratory since 1963. A production-scale facility for converting HLLW to glass, under construction at Marcoule, France, is scheduled to begin operation in 1977. Several alternative methods for converting HLLW to a glass are being developed in the U.S. within ERDA programs.

NON-HIGH-LEVEL LIQUID WASTES

This classification includes a wide variety of waste types. Some plant wastes, such as laundry and decontamination solutions, are common. But the category also includes wastes specific to certain plants, such as acid etch solutions from a mixed-oxide fuel fabrication plant or alkaline solvent wash solutions from a fuel reprocessing plant. Thus the non-high-level liquid waste category encompasses a broad spectrum of radioactive liquid wastes varying widely in chemical and radioactive content.

Because of the diversity of non-high-level liquid waste types and because the goal is usually to clean large volumes so that the water can be released as a plant effluent, a sequence of treatment processes is usually required. The treatments are relatively conventional technology. In addition to evaporation, which is the most commonly used treatment technology, treatments can include reverse osmosis, flocculation, precipitation, filtration, and ion exchange. The technologies are all available, and most are currently being used in the nuclear industry.

Certain non-high-level liquid wastes, such as waste solvent from a reprocessing facility, are flammable. Technology is available for the incineration of these wastes.

IMMOBILIZATION OF NON-HIGH-LEVEL WASTES

Virtually all nuclear fuel cycle operations generate liquid or solid wastes requiring immobilization prior to transportation and storage. Many alternative technologies are currently in use to immobilize these wastes. The alternative technologies include the use of absorbents and the incorporation of the wastes in concrete, asphalt or urea-formaldehyde resin.

Absorbents have been used widely in both the chemical and nuclear industry to immobilize liquids for transportation and disposal. Some of the typical granular or powdered absorbent materials include vermiculite, silica gels, plaster of paris, and various clays. The absorbent method, properly applied, will entrap the waste liquid so that no free liquid exists within the bulk material.

Incorporation of radioactive wastes in cement or concrete has been a common practice for many years. The optimum proportions of cement and waste vary with the type of waste to be solidified. Maximum waste contents for the solidified products are typically 75 wt% for solid waste or 33 wt% for aqueous solutions or slurries.

Commercialized technology is available for the immobilization of a wide variety of wastes in asphalt, or bitumen, including neutralized evaporator concentrates, sludges, ion exchange media, and incinerator ashes. Both batch and continuous processes are in use, particularly in Europe.

Systems for immobilizing reactor wastes in urea-formaldehyde have been marketed commercially since 1972.

COMBUSTIBLE SOLID WASTES

Combustible solid radioactive wastes include a large variety of items such as paper, rags, plastic sheeting, protective clothing, gloves, rubber shoes, wood, organic ion exchange resins, filter aids, etc. Much of the waste material is collected as general trash, which usually consists of a mixture of combustible and noncombustible items. Therefore, treatment options generally include sorting prior to treatment of the wastes.

Waste compositions vary depending on the operations involved. General trash, or non-glove box waste, includes mainly cellulosic materials such as paper, wood, cardboard, absorbent cotton, cotton clothing, and rags. Usually these materials burn readily to release water vapor and carbon dioxide. Glove box-generated wastes, on the other hand, contain chiefly rubber or plastic materials since cellulosic materials are mostly excluded from glove box operations.

Various treatment technologies are available for combustible wastes. They range from simple storage in boxes or steel drums to compaction, encapsulation in concrete, and incineration.

Compaction or baling is a mechanical operation for reducing the volume of the waste. For a typical combustible waste, a volume reduction of 5- to 10-fold can be expected. Shredding of the waste before compacting is sometimes done to increase the volume reduction ratio.

Some combustible wastes have been immobilized by mixing with concrete. An example of this type of operation is the incorporation of dewatered ion exchange resin in cement.

The incineration of combustible waste is simple in principle but requires special considerations when applied to radioactive materials. The most important of these considerations have been associated with the

off-gas cleanup system. A number of alternative incineration systems are being developed and evaluated in the U.S. These systems include simple single-chamber units, fluidized beds, moving grates, and rotary kilns. The operational modes include excess air incineration, controlled air incineration, and acid digestion.

NONCOMBUSTIBLE SOLID WASTES

The primary constituent of these solid wastes is metal, but other noncombustibles such as glass and concrete are also present. Incidental quantities of combustible material such as grease, plastic, and floor sweepings may also be present with the noncombustible waste.

Examples of items that become noncombustible wastes are inoperable and obsolete process equipment such as pumps, process vessels, piping, and laboratory or process glove boxes. These large items present a unique waste management problem due to size and, in some cases, high radiation levels. Small items also in this waste category include filters, broken glassware, and shielding material.

The smaller noncombustible solid waste items are often associated with combustible materials. As an example, plant maintenance activities such as replacement of a process valve will contaminate plastic, rags, and other combustible items, thus requiring a prepackaging sorting or segregation operation.

The steps in management of large noncombustible waste items are decontamination, disassembling, and packaging. Commercialized technology is available for these steps. Decontamination usually includes chemical flushing with several different reagents but may also include mechanical activities. Disassembling may be done by sawing or mechanical cutting or tearing, as well as cutting with torches or electric arc. Packaging can be done either with or without crushing or mechanical compaction.

Advanced alternatives being developed for the future are directed toward decreasing the final volume of these wastes and decreasing the individual mechanical operations required in their treatment. These alternatives include melting and dissolution.

GASEOUS WASTES

Gaseous wastes include radioactive noble gases, iodine, tritium and carbon-14. Treatment technology is available for the noble gases and iodine, and is being developed for tritium and carbon-14. The technologies being developed for tritium and carbon-14 are to meet a possible future need; their present release rates do not require treatment under present regulations.

Two technologies are available to collect noble gases from off-gas effluents. One is the cryogenic treatment of the entire off-gas stream followed by distillation of the resultant liquified air for purification of the noble gas fraction. This technology is currently used at the Idaho Chemical Processing Plant (ICPP). The second technique is collection by liquid fluorocarbon absorption. This technology is presently being offered commercially for capture of noble gases released from light water reactors.

At present, captured noble gases are stored in pressurized gas cylinders. Development is currently underway on encapsulating noble gases in a sodalite-zeolite matrix. The advantage of the zeolite encapsulation is that long-term storage of noble gases does not depend on the integrity of pressure vessels.

Iodine can be removed from effluent off-gas streams by using one of many available technologies which include caustic scrubbing, chemisorption by metal-loaded absorbents, mercuric nitrate-nitric acid scrubbing, and iodox scrubbing. The choice of technique depends on the nature of the off-gas stream to be treated and the desired final product form of the captured iodine.

DECONTAMINATION AND DECOMMISSIONING OF RETIRED FACILITIES

Facilities contaminated in fuel cycle operations can become a major waste quantity upon retirement. Three alternatives for decommissioning contaminated facilities have been identified in Regulatory Guide 1.86 of the Nuclear Regulatory Commission. The alternatives are protective storage, entombment, and dismantling.

Protective storage is often referred to as mothballing. In this decommissioning alternative the facility is prepared to be left in place safely for an extended period which might range from decades to two or three centuries. Potentially mobile radioactive materials are removed from the site. All operational systems and support utilities are placed in a nonoperation mode. A continuous surveillance program is established.

Entombment involves all the decommissioning steps of protective storage but in addition provides for sealing all contaminated facility components in a high-integrity structure. Such contaminated components might include the pressure vessel and internals of a light water reactor or the major processing vessels of a reprocessing facility.

The dismantling alternative involves removing from the facility site all radioactive components and materials which exceed the criteria for unrestricted release. Once all the radioactivity is gone, all restrictions on the site are removed. This mode of decommissioning may be applied

immediately after facility retirement or following a period of protective storage to allow decay of short-lived radionuclides.

All three of the decommissioning alternatives have been employed. For example, four reactors at Hanford and the Fermi reactor in Michigan have been placed in protective storage and are currently in this status. The Hallam nuclear power facility near Hallam, Nebraska, has been entombed. After shutdown in 1968 the reactor at Elk River, Minnesota, was dismantled and the reactor site was returned to unrestricted use. Other facilities have also been subjected to these decommissioning technologies.

INTERIM STORAGE OF WASTE

There are several points during the operational sequence of an integrated waste management program at which interim storage should or can be provided for various waste forms. The interim storage may be a holding, or surge, point between operations, as is the case with interim liquid storage of both high-level and intermediate-level wastes. Interim storage may also be a holding point for final solid waste products prior to final isolation or disposal.

Interim storage facilities are designed to minimize radiation exposures to man and to provide confinement of the waste for as long as necessary. At the end of the interim storage period for solid wastes, the waste containers must be capable of being retrieved and transported to future treatment or storage sites, preferably without need for repackaging. Consideration must also be given to protecting the waste from accidental intrusion, vandalism, and other adverse acts.

Interim storage has been used for wastes representative of all of the wastes from the back end of the LWR fuel cycle, either commercially or in ERDA facilities. The storage technologies include tank storage of liquid wastes, water basin storage of spent fuel elements, and warehouse and other engineered surface storage of solid wastes. Retrievable surface storage of solidified high-level wastes has been investigated but not implemented.

TRANSPORTATION

Transportation of nuclear material is required between operating facilities in the nuclear fuel cycle. These transportation steps include movement of spent nuclear fuel from the reactors to reprocessing facilities, movement of nuclear waste from a reprocessing facility to a Federal repository, and movement of recovered plutonium, all of which must be done within the Department of Transportation regulations.

Transportation of Spent Fuel

The technology and hardware to transport spent light water reactor fuel is in use. Truck casks and rail casks are presently available. Most of

these casks will accept PWR or BWR spent fuel with a slight change of the casks internals.

Transportation of Waste Forms

The nuclear waste forms that require transportation include solidified high-level waste, chop-leach fuel bundle residues, and non-high-level solid wastes.

To date no solidified high-level waste has been generated by the commercial sector of the industry; therefore, none has been shipped. Conceptual designs are available for heavily shielded shipping casks to transport this waste form. These casks will be similar to currently available spent fuel casks.

Also to date, no chop-leach fuel bundle residues have been required to be shipped offsite. Thus no shipping containers specifically designed for these wastes are presently available. The wastes can, however, be shipped using conventional radioactive shipping container technology.

Non-high-level waste is generated at all back end fuel cycle facilities. To accommodate the various waste forms in this category, several waste transportation containers and casks have been developed, licensed, and are commercially available today. Special non-high-level waste forms may be generated that require new casks, but these new casks can be developed with existing technology.

Plutonium is separated from spent fuel at a reprocessing facility. The separated plutonium is then shipped to a fuel fabrication plant. By regulation, all plutonium shipped after June 1978 will be in a solid form. Small quantities of plutonium are now being shipped routinely. Advanced plutonium cask designs are presently being developed to accommodate the larger quantities of plutonium that will require transportation in a mature industry. These developments would also be available for wastes containing plutonium.

FINAL ISOLATION AND DISPOSAL

There are two basic alternatives for final management of wastes generated by the commercial LWR fuel cycle: 1) geologic isolation and 2) elimination from existence on earth. Geologic isolation, that is, placement in geologic formations, offers two options: storage, which provides for retrieval of wastes after emplacement, and disposal, which uses emplacement techniques based on anticipation that the wastes would not be retrieved.

Two elimination alternatives have been defined: extraterrestrial ejection by rocket, and transmutation, in which nuclides are bombarded by nuclear particles or radiation with the objective of efficiently

converting the bombarded nuclides to new species that will decay more quickly and/or emit less hazardous radiation.

The elimination options can be used only for selected waste components. The separations technology needed has not been developed.

The fundamental concept for geologic isolation is emplacement of wastes in geologic media expected, because of their stability, to maintain isolation for as long as necessary. Geologic isolation could be further augmented if the geologic medium has capability to retard nuclide migration.

Four basic geologic environments have been identified as candidates for isolation of radioactive wastes: the sea floor, ice sheets, formations under the continents at depths isolated from the biosphere, and shallow continental geologic formations used as burial grounds.

With present technology, opportunities to use these geologic isolation alternatives can be summarized as follows:

- . For ice sheets, additional geologic and meteorologic data are needed.
- . Disposal in geologic formations in the sea floor is being investigated. The technical information base is at present insufficient, but all information available to date is supportive of the concept.
- . Emplacement in burial grounds, i.e., shallow geologic formations, has been used in the past and is anticipated for the future. Past use of this concept has not fully utilized a systematic approach to isolation in shallow geologic formations; practice is being improved.
- . Numerous concepts for disposal in deep geologic formation (i.e., emplacement without anticipation or retrieval) have been defined. Most concepts are feasible with current technology, but long-term post-emplacement stability has not been evaluated.
- . Deep continental geologic formations suitable for disposal or permanent storage have been identified. Procedures and technology for selecting specific repository sites are available, and conventional engineering practice can be used to design repository systems.

Geologic storage or disposal relies on the geologic medium to serve as a stable barrier that keeps the wastes isolated. The geologic barrier can be augmented by selecting waste forms and engineered structures that contribute to the stability of the system and, in storage systems, provide capability for waste retrieval.

Design of geologic isolation repositories is specific to the site, geologic medium, and characteristics of the wastes to be emplaced. Factors considered in the design of waste handling facilities and emplacement configurations include size and shape of the waste containers; heat and

radiation emissions from the waste; chemical, thermal, mechanical, and radiation interactions between the waste and the geologic medium; and requirements for operational safety. Repositories can be excavated using conventional mining technology.

Repositories can be initially designed for "provisional" geologic storage; i.e., they can be designed for waste retrieval using methods and rates essentially the same as those used for waste emplacement. Such repositories can also, after a period of operation during which data are taken and evaluations are made to test long-term stability, be converted to "permanent" geologic storage by backfilling and sealing.

1.0 ALTERNATIVES FOR THE BACK END OF THE LWR FUEL CYCLE

1.0 ALTERNATIVES FOR THE BACK END OF THE LWR FUEL CYCLE

The LWR fuel cycle can be operated to include recovery of uranium and plutonium from spent fuel so that these energy resources can be formed into new fuel elements and recycled to the power reactors. The back end of the fuel cycle can also be operated so that neither of these materials is recovered and recycled or so that either of them is recycled. Additional options can be defined when alternative technologies for fuel cycle operations and waste management are considered.

There are, therefore, numerous alternatives for the back end of the LWR fuel cycle. The technologies used and the materials recycle scheme that is selected will influence the quantities and types of wastes generated. Holdup times (e.g., storage time for spent fuel prior to reprocessing) will also influence waste quantities to be managed at any time and logistical factors such as capacity of waste storage facilities. Consequently, waste management requirements are highly dependent on how the back end of the fuel cycle is operated.

This section provides an overview characterization of waste management for three representative fuel cycle options. In each case, use is made of available waste management technology. Technologies and operating modes that differ from those used for these examples would be expected to produce different waste quantities and waste management logistics.

1.1 FUEL CYCLE OPTIONS

The three basic fuel cycle operating modes considered are:

Mode I. No Recycle of Uranium or Plutonium. This fuel cycle option is often referred to as the "throwaway" fuel cycle. In this option neither uranium or plutonium is recycled. The spent fuel would be managed as a waste.

Mode II. Uranium Recycle Only. In this version of the fuel cycle, spent fuel would be reprocessed to separate uranium and plutonium from the fission products, but only the uranium would be recycled. The plutonium oxide product from the reprocessing facility would be placed in storage. The uranium hexafluoride product from the reprocessing operation would be sent to an enrichment facility so that the enriched uranium could be fabricated into uranium oxide fuel and recycled back to light water reactors.

Mode III. Recycle of Plutonium and Uranium. In this full recycle option the spent nuclear fuel, following a brief interim storage period at the reactor, is transported to a nuclear fuels reprocessing facility. Here the spent fuel is processed into three major fractions: the high-level waste, the uranium product, and the plutonium product. The uranium product is converted to uranium hexafluoride, and the plutonium is converted to plutonium oxide. The uranium product is introduced to a uranium enrichment facility. The plutonium

oxide is sent to a fabrication plant, where it is blended with natural uranium and fabricated into plutonium and uranium mixed-oxide fuel. The resultant enriched uranium from the enrichment facility is sent to the fabrication facility, where it is fabricated into uranium oxide fuel, and subsequently both the mixed-oxide fuel and the uranium oxide fuel are both recycled to light water reactors. The high-level waste is solidified by the commercial reprocessor and given over to Federal custody for management.

1.2 WASTE MANAGEMENT FOR THE FUEL CYCLE OPTIONS

The three fuel cycle operating modes will in part yield similar wastes. In particular, reactor operations will not differ; reactor wastes will therefore be essentially the same for each fuel cycle mode. There will, however, be differences in radiological composition, physical characteristics and quantity of some of the other fuel cycle wastes.

The differences in radiological composition result because in Mode I, the "throwaway" fuel cycle, all of the uranium and plutonium becomes waste. About 200 times the amount of uranium and plutonium normally expected in the high-level waste from fuel cycle Mode III, full recycle operation, would be discarded with the spent fuel assemblies. On the other hand, the nonplutonium transuranic content of the Mode III wastes will be higher than in the modes in which plutonium is not recycled. This means, for instance, that the neutron emission rate from the Mode III high-level waste is higher than from Mode II high-level waste. Details of the differences in radiological composition of the wastes from the three fuel cycle operating modes are given in Tables 2.17 to 2.22 in Section 2.4.3.

1.2.1 Waste Management Systems for the Fuel Cycle Options

Figures 1.1 to 1.5 show waste management systems that can be established for each of the three fuel cycle operating modes using available technology described in this document. As previously noted, the systems shown are examples. The systems shown are feasible and representative of current technology, but there is no intent to imply that they are optimum systems for the fuel cycle options.

Mode I. Figure 1.1 shows a reference waste management system for Mode I, or the "throwaway" fuel cycle. For this operating mode the only operating components in the back end of the fuel cycle are the reactors and one or more interim storage repositories for the spent fuel assemblies. Each such repository, consisting of a water basin, packaging and possible final storage or disposal facilities, would service a large number of reactors. The reactor operators would ship their spent fuel directly to these facilities instead of to a reprocessing plant.

At the repository the fuel would be stored for several years in a water basin to allow decay in the radioactive heat generation rate sufficient to

FIGURE 1.1. Origin, Treatment and Disposition of Radioactive Wastes from a Power Reactor (Mode I)

minimize heat removal problems in subsequent final storage or disposal. Wastes resulting from operation of water basin storage facilities are discussed in Section 2.3. Techniques for packaging the spent fuel elements are described in Section 17.3. For this characterization, it was assumed that the fuel assemblies would be packaged in mild-steel containers in groups of four, surrounded by a high-thermal-conductivity metal filler to maintain the fuel temperatures as low as possible. It also was assumed that the fuel would be stored in engineered surface or near-surface facilities using passive cooling. Such storage procedures are described in Section 17.4. The fuel containers would subsequently be transferred to deep geologic storage.

The reactor operating wastes are treated in a conventional manner as shown in Figure 1.1. The wet sludges, resins, etc., would be incorporated in concrete and the dry wastes would be compacted and packaged before shipment to shallow geologic isolation (burial grounds). The waste volumes vary significantly, depending on the reactor type and the method of condensate treatment. A detailed description of reactor wastes is given in Section 2.2.

Mode II. In Mode II fuel cycle operation the spent fuel is reprocessed and the uranium is recycled through enrichment and fuel fabrication. Plutonium, considered to be a waste or byproduct in this operating mode, would be converted to oxide and sent to a storage facility. Figures 1.2 and 1.3 show reference waste management systems for the two back end-of-the-fuel-cycle components of Mode II operation, i.e., the reactors and fuel reprocessing plants.

Wastes from reactors would be managed using methods the same as those described for Mode I.

Operation of the reprocessing plant generates a diverse mixture of wastes, as shown in Figure 1.3. This figure shows the major reprocessing plant sources for the various waste categories and the treatment methods for these wastes. More details on these wastes and their compositions are given in Section 2.4. If the waste treatment generates secondary wastes, as is the case, for example, for vitrification of high-level waste, the volume of the secondary wastes has been factored into the overall waste volumes. About 70% by volume of the reprocessing plant wastes are TRU-contaminated; it is assumed that these wastes will ultimately be placed in deep geologic storage repositories.

Iodine was assumed to be trapped and removed from the off-gas, but noble gases, tritium and carbon-14 were assumed to be released in the gaseous plant effluents, as is current practice. Process water was assumed to be vaporized and released as part of the gaseous plant effluents.

FIGURE 1.2. Origin, Treatment and Disposition of Radioactive Wastes from a Power Reactor (Modes II and III)

FIGURE 1.3. Origin, Treatment and Disposition of Radioactive Wastes from a Reprocessing Plant (Mode II)

The plutonium was assumed to be stored in about 5-kg batches in steel containers designed to prevent criticality. As in the case of spent fuel elements, interim engineered storage was assumed, to be followed, if appropriate, by deep geologic storage.

Mode III. As shown in Figure 1.4, the reprocessing plant wastes for Mode III operation are identical to those of Mode II operation except that the PuO_2 product is now shipped to a mixed-oxide (MOX) fuel fabrication plant instead of being packaged and shipped to a Federal repository. Mode III operation generates additional wastes because of MOX fuel fabrication.

Figure 1.5 shows the major sources of wastes in a MOX fuel fabrication plant. Further information on these wastes is given in Section 2.5. All of the operating wastes from MOX fuel fabrication plants must be considered potentially TRU-contaminated, and are therefore shown being sent ultimately to deep geologic isolation or the equivalent.

1.2.2 Impact of Fuel Cycle Options on Waste Generation

The quantities and characteristics of wastes generated in the back end of the LWR fuel cycle are, as previously noted, sensitive to the technologies and logistics of the fuel cycle. This section outlines the impact of fuel cycle modes on waste quantities and characteristics.

The three fuel cycle modes described in Section 1.2.1 were used to compare alternatives for the back end of the LWR fuel cycle. To describe the operating modes completely it is also necessary to define the cooling times, i.e., interim storage periods, at various stages in the cycle. In practice, the cooling times are expected to vary somewhat, and these variations can have important effects on waste characteristics. The cooling times used can even affect the selection of specific waste management technologies.

Although cooling times are an important logistics parameter, a thorough analysis of the effects of all of the possible combinations is beyond the scope of this document. One set of representative cooling times was selected for each operating mode, as shown in Table 1.1.

TABLE 1.1 Reference Interim Storage Periods for the Three Fuel Cycle Modes

<u>Case</u>	<u>Spent Fuel Cooling Period, years</u>	<u>Extent of Recycle</u>	<u>HLLW Cooling Period, years</u>	<u>Interim Solid Waste Cooling Period, years</u>	<u>Interim Hulls Cooling Period, years</u>
Mode I		None	--	--	--
Mode II	5	U	1	5	6
Mode III	5	U, Pu	1	5	6

FIGURE 1.4. Origin, Treatment and Disposition of Radioactive Wastes from a Reprocessing Plant (Mode III)

FIGURE 1.5. Origin, Treatment and Disposition of Radioactive Wastes from a MOX Fuel Fabrication Plant (Mode III)

None of these cases will necessarily describe the LWR fuel cycle as it finally evolves since the spent fuel cooling time, recycle configuration, and waste management technology will change with experience and with the development of commercial reprocessing and waste treatment technology. This approach to the case description was selected because it allows separation of key variables in order to test their impacts on waste generation rates, design of facilities, and selection of final storage or disposal concepts.

Methodology. These characterizations were based on the moderate electrical (low nuclear growth projections in WASH-1139-(75). Projections of waste quantities are not the same as those given in Section 3 of this report primarily because of differences in assumed fuel burnups. The range of projections given here and in Section 3 is expected to span the alternatives for LWR fuel cycle operations; use of Section 3 values in these analyses would reflect expected alternative limits for waste quantities and characteristics.

The computer code NUFUEL⁽¹⁾ was used to generate the mass flows and material inventories in the LWR fuel cycle for the years 1975 through 2000 for each case analyzed. The mass flows from NUFUEL and waste generation coefficients from fuel cycle component diagrams (Figures 1.1 to 1.5) were used to accumulate waste inventories through year 1999 and to calculate flows in the waste streams in the year 2000. The various waste streams were categorized as follows: TRU-Shielded and Cooled (High-Level Waste), TRU-Shielded (but not cooled), Tru (not shielded or cooled), Hulls, and Non-TRU. These categories are shown in parenthesis on each component diagram. The non-TRU wastes are expected to be placed in shallow geologic isolation (burial grounds). All TRU waste forms are expected to be placed in deep geologic isolation.

The ORIGEN computer code⁽²⁾ was used to calculate the radiological properties of the spent fuel and high-level and hulls waste streams for each fuel cycle configuration. Low-exposure first- and second-discharge fuel was treated explicitly for each reactor in an attempt to accurately quantify the spent fuel compositions during this LWR growth period. First-recycle plutonium was used for LWR mixed oxide fuel since it was observed from the mass flows that the breeder reactor system utilized sufficient LWR plutonium to preclude significant amounts of multiple recycle in LWRs prior to year 2000.

The ORIGEN files were accumulated and decayed to represent the radiological properties of spent fuel and waste inventories through the year 1999 and mass flows of these materials in year 2000. The inventories and properties of the spent fuel waste streams thus calculated were used in the case comparisons given in the following sections.

Impact of Recycle Options

Modes I, II and III are equivalent with respect to fuel cycle delay times but vary with respect to fuel materials recycle. These variations were investigated to determine the effect of recycle fuels on the types and quantities of wastes from the LWR fuel cycle.

Implications of Restricted Recycle. In Modes II and III, spent fuel is reprocessed, and uranium and plutonium (Mode III) or uranium only (Mode II) are recycled to decrease the amount of fresh uranium required to fuel the LWR economy. In Mode I no fuel materials recycle is employed. The effect on uranium demand is shown in Figures 1.6, 1.7 and 1.8 which project mass flows through 1999 and in the Year 2000. The total U_3O_8 demands through Year 2000 are 1.95, 1.79 and 1.54 million short tons of U_3O_8 for Modes I, II and III respectively. Mode I does not provide separated plutonium; it therefore precludes development of the fast breeder reactor.

Waste Volumes. The waste inventories for each of the recycle modes are shown in Figures 1.6, 1.7 and 1.8. There are no significant differences in waste volumes for Modes II and III; in Mode I several effects of operation are apparent. Engineered storage of packaged spent fuel elements is required before deep geologic isolation. The volume of packaged spent fuel is significantly less than the volume of TRU wastes from reprocessing and mixed oxide fabrication. However, the volume of non-TRU wastes is only slightly less than with Modes II and III because the reactor is the primary source of this waste category.

Materials Processed in Year 2000. To highlight differences between the fuel recycle modes, certain indices were selected to characterize the reprocessed spent fuel and the high-level liquid waste solidified during the Year 2000. Specific activity (curies/unit throughput) and neutron source strength (neutrons/sec/unit throughput) were selected to compare the degree of shielding required during transport and processing steps. The heat load (watts/unit throughput) was selected to compare cooling requirements, and the toxicity index per unit throughput was selected as a first-order indicator of the relative risk involved in handling and transporting these materials. For comparison, similar indices were calculated for spent fuel packaged in Mode I since no reprocessing or solidification takes place in this fuel cycle configuration.

In Figure 1.9, Modes I, II, and III are compared in terms of materials processed in the Year 2000. The differences between Modes II and III are due to higher concentrations of transuranics in plutonium recycle fuels. Mode I is significantly different due to the 10-year cooling period before packaging as compared to 5-year cooling before reprocessing in Modes II and III.

FIGURE 1.6. Mode I - 10 Year Spent Fuel Cooling, Then Engineered Storage

FIGURE 1.7. Mode II - 5 Year Spent Fuel Cooling, 1 Year HLLW Cooling - U Recycle Only

FIGURE 1.8. Mode III - 5 Year Spent Fuel Cooling, 1 Year HLLW Cooling

FIGURE 1.9. Modes Compared in Terms of Materials Processed in the Year 2000

Utilization of Plutonium

Distributions of fissile plutonium at the beginning of the Year 2000 for Modes I, II and III are given in Table 1.2. In Mode I, all the plutonium remains in spent fuel form in residues in the spent fuel cooling facilities, or in engineered storage residues. In Mode II, roughly half of the plutonium recovered from reprocessing residues is placed in plutonium storage, the balance being used for fast-breeder reactors. In Mode III, where plutonium is recycled, about 111.6 MT (12.7%) more plutonium than used in Mode II is ultimately utilized in the LWR system. Mixed oxide fuel fabrication from LWRs produces more plutonium for deep geologic inventories than do high-level wastes. Plutonium-bearing wastes from LMFBF fabrication and reprocessing are not shown in these comparisons.

TABLE 1.2 Distribution of Fissile Plutonium, MT - By^(a) 2000

	<u>Mode I</u>	<u>Mode II</u>	<u>Mode III</u>
In Spent Fuel Inventories	625.9	373.1	431.7
To LWR	0	0	273.2
To FBR	0	242.5	242.5
In Pu Storage	--	260.9	36.5
In HLLW	--	0.3	0.3
In Solidified HLW	253.0 ^(b)	2.3	2.4
In TRU Wastes from LWR MOX Fab.	--	0	4.1

a. Beginning-of-year

b. Packaged Spent Fuel in Engineered Storage

RADIOACTIVE AIRBORNE RELEASES

Airborne releases from reactors are the same for each fuel cycle mode. There are, however, significant gaseous releases during reprocessing, and these releases therefore differ for Mode I in comparison with Modes II and III. Table 1.3 lists the airborne releases from reprocessing for Mode III; since these gases are produced via fission and activation, the corresponding quantities for Mode II are expected to be approximately the same. In Mode I, however, reprocessing does not occur and these gases are simply packaged with the spent fuel and sent to the engineered storage facility.

TABLE 1.3 Gases Released During Reprocessing - Mode III

	<u>Mass, grams</u>	<u>Ci</u>	<u>Toxicity Index, m³ of water</u>
Tritium	2.24×10^3	2.18×10^7	7.27×10^9
Carbon	6.09×10^6	2.71×10^5	3.39×10^8
Krypton	2.35×10^7	5.38×10^8	--
Iodine ^(a)	2.06×10^3	2.86	4.76×10^7
Xenon	4.09×10^8	--	--

a. Assumes Iodine decontamination factor of 10^{-3} .

REFERENCES

1. Computer Program NUFUEL for Forecasting Nuclear Fuel Requirements and Related Quantities, WASH-1348, USAEC, October, 1974.
2. M. J. Bell, ORIGEN--The ORNL Isotope Generation and Depletion Code, ORNL-4628, Oak Ridge National Laboratory, Oak Ridge, TN, May 1973.

2.0 LWR FUEL CYCLE WASTES

2.0 LWR FUEL CYCLE WASTES

This section describes the LWR fuel cycle and the wastes it can generate. There are several fuel cycle alternatives ranging from no recycle of uranium and plutonium in spent fuel to maximum recycle of both of these fuel material resources. Waste generation will in practice depend on recycle mode, fuel cycle operations technology and logistics, and waste management technology. The descriptions of the fuel cycle and waste types given in this section encompass the broadest range of operating conditions and waste management consequences expected for the back end of the LWR fuel cycle.

2.1 DESCRIPTION OF THE LIGHT WATER REACTOR (LWR) FUEL CYCLE⁽¹⁾

The LWR fuel cycle is the system of operations and facilities for the preparation, utilization, and reconstitution of LWR fuel. The main operations involve 1) mining of ore, 2) milling, i.e., ore concentration and purification, 3) conversion of U_3O_8 to UF_6 , 4) enrichment, 5) conversion of enriched UF_6 to UO_2 , 6) fuel fabrication, 7) irradiation in a reactor, and 8) reprocessing. Transportation of materials between the facilities where these operations are carried out and collection and storage of wastes are also considered parts of the fuel cycle. The fuel cycle is illustrated in the diagram of Figure 2.1. Closing the fuel cycle, that is, providing for operation of reprocessing plants and recycle of plutonium into LWR fuel as mixed oxides of plutonium and uranium (MOX), increases the uranium utilization by recovery of unused fissile ^{235}U and the fissile plutonium formed from ^{238}U .

2.1.1 Mining⁽¹⁻³⁾

A major portion of domestic uranium ore is found in New Mexico, Wyoming, Colorado, and Utah. Domestic uranium mines have the capacity to produce about 8 million metric tons* (MT) of ore per year. The assay of uranium ore averages about 0.2% or 2 kg of U_3O_8 per MT of ore. Two methods, open pit mining and underground mining, are used to produce most of the uranium in the United States.

Underground mining typically involves service buildings, hoisting and truck loading facilities, a mine waste pile, a flow of water pumped to surface drainage from underground sumps, shops, warehouse, changehouse, and office. Underground openings may extend laterally to a mile or so. Solid wastes (primarily nonradioactive) may accumulate in volumes equal to the volume of processed ore.

Open pit mining, on the other hand, typically involves a large open excavation, large piles of earth and rock overburden resulting from the excavation, a network of operating roads and yards, and a flow of mine water pumped into the local surface drainage. The site may include shops, warehouse,

*A metric ton is 1000 kg, or 2204.6 lb.

office, changehouse, and an assortment of heavy earth-moving equipment. The ore body may lie at various levels from 30 to 140 meters (m) below the surface in readily friable rock strata. The stripping ratio, that is, the volume of overburden to volume of ore body, may average about 30 to 1. Wastes produced by mining are distributed in gaseous, solid, and liquid materials.

Airborne radioactive effluents such as uranium-bearing dusts and radon and its decay products are released to the atmosphere when an ore body is exposed and broken up during mining operations. The concentrations of airborne

FIGURE 2.1. The LWR Fuel Cycle⁽⁴⁾

radionuclides in unrestricted areas, at least 600 m from the point of release, however, are expected to be below the limit of detection.

The primary solid waste material produced by mining is the barren rock and earth overburden, most of which may ultimately be disposed of as backfill. The small quantity of uranium and its decay products originally contained in the earth overburden may thus be returned to the ground as part of the back-fill material.

The largest liquid waste stream from open pit mining operations is mine drainage water. Mine drainage water frequently carries suspended solids, silicates, and trace uranium ore from the pit area. Dissolved and suspended uranium and its decay products, equivalent to several curies per 1000 MWe-year,* are dissolved and suspended in mine drainage water.⁽¹⁾ The wastewater is treated in settling pond operations by seepage and evaporation as it recycles into the hydrosphere.

Projections of the amounts of ore mined and mine tailings produced for U. S. LWRs are listed in Section 3.2.

2.1.2 Milling^(1-3, 5)

A uranium mill extracts uranium from the ore by mechanical and chemical processing of the mineral into a semi-refined product, commonly called "yellowcake," which contains 70 to 90% U_3O_8 . About 20 mills are operating in the United States with a total annual capacity of 17,000 MT of U_3O_8 . Uranium mills are usually located near the source of ore; typically a uranium mill is located in an arid, isolated region of the western states.

Two methods are employed for uranium milling operations in the U. S.: the acid and the sodium carbonate leach process. The choice depends upon the characteristics of the ore being processed. Most of the ores are suited to the acid leach process which produces about 80% of all yellowcake. The basic steps in the acid leach process include:

1. Ores are crushed and blended, then wet ground in a rod or ball mill, and transferred as a slurry to leaching tanks.
2. The ore is contacted with sulfuric acid solution and an oxidizing reagent to leach the uranium from the ore. The residues (tailings) are washed, and the product liquor is pumped to solvent-extraction units.
3. In the solvent-extraction step, the uranium is purified and concentrated.

*1000 MWe-year is the power generated by a reactor with a capacity of 1000 MWe, operating for a calendar year. Since, in actual practice, reactors are not operated continuously, a time of 1.25 to 1.55 years might be required for a reactor of 1000 MWe capacity to produce the power of 1000 MWe-year.

2.4

4. The uranium is precipitated with ammonia and transferred as a slurry.
 5. The concentrate is centrifuged to thicken and separate it from residual liquids.
 6. The concentrate is calcined and pulverized.
 7. The powdered yellowcake is packaged in 0.2-m³ drums for shipment.
- The milling operation produces wastes in gaseous, liquid, and solid form.

The gaseous medium is air which normally carries radioactive gases and particulate solids away from the operation. Uranium and uranium decay products (²³⁰Th and ²²⁶Ra) are released from ore piles, the tailing retention system, and the ore crushing-grinding ventilation system in the form of dust. Natural uranium is released from the yellowcake drying and packaging operations as entrained solids. Radon gas is released from the leach tank vents, ore piles, tailings retention system, and the pulverization exhaust system. The concentrations of airborne radioactive particulate solids released into the environment from processing operations are limited by dust collection systems.

Liquid wastes from mills consist of solutions containing spent chemicals from the grinding, leaching, and washing operations. Since concentrations of ²²⁶Ra and ²³⁰Th may be above the limits specified by regulation, dispersal of this material to the environment is prevented. The neutralized waste liquor is stored in a tailings retention pond to prevent discharge into the surface water system and to minimize percolation into the ground.

Solid wastes (tailings) from mills are composed mostly of sandstone and clay particles. They contain traces of waste chemicals, and mine waste. These solid waste tailings may contain about 85% of the radioactive materials originally in the ore. The management of these tailings piles has received increased attention to limit the local inhalation of gaseous ²²²Rn, the radioactive decay product of ²²⁶Ra, to prevent the fines of these tailings from becoming airborne, and to prevent the runoff of tailings into surface streams.

2.1.3 Uranium Hexafluoride Production and Uranium Enrichment^(1,3,6)

The U₃O₈ concentrate extracted from the ore must be converted to the volatile compound uranium hexafluoride (UF₆) for enrichment by the gaseous diffusion process. Two processes differing only in the head-end are used for UF₆ production. Roughly equal quantities of UF₆ feed to the enrichment plants are produced by each method. The dry hydrofluor process results in retention of impurities in either volatile compounds or as solid ash. The wet solvent extraction method separates impurities by extracting the uranium values in an organic solvent, leaving other constituents in the aqueous phase.

The dry hydrofluor process consists of these typical operations:

1. pre-process handling, weighing, sampling, and storage
2. fluidized-bed reduction of the U_3O_8 to UO_2 by cracked ammonia
3. fluidized-bed hydrofluorination with anhydrous HF to form a crude UF_4
4. fluidized-bed fluorination with elemental fluorine to form crude UF_6
5. fractional distillation to form refined UF_6 .

In this process the impurities are separated from the product by taking advantage of relative differences in volatility.

The wet solvent extraction method consists of the following operations:

1. pre-process handling, weighing, sampling, and storage
2. digestion in hot nitric acid
3. countercurrent solvent extraction with tributylphosphate (TBP) in hexane
4. stripping of uranium as uranyl nitrate into an aqueous solution
5. calcining to UO_3
6. fluidized-bed reduction to UO_2 with cracked ammonia
7. hydrofluorination in a two stage countercurrent reactor to UF_4 using anhydrous HF
8. fluorination to UF_6 by reaction with elemental fluorine.

The two methods differ mainly in that the wet process starts with a solvent extraction step to purify the ore concentrate before fluorination, whereas the dry process converts the concentrate to UF_6 in fluidized-bed reactor stages and as a final step purifies the UF_6 by fractional distillation.

The wastes from the two methods of UF_6 production differ substantially. Hydrofluor process wastes appear in both the gaseous and the solid states, whereas wet solvent extraction process wastes are primarily liquids.

The more contaminated liquid wastes, the raffinate from the solvent extraction process, are held in sealed ponds; only treated coolant and dilute scrubber solutions combined with all other clear water effluents are released. The 12-month average concentration of radioactive materials measured in the raffinate may be about 10% of the limits for natural uranium and $^{226}Ra^*$ in water effluents released to an unrestricted area. The raffinate liquid is held indefinitely in ponds; it is not directly released to the environment, but about 1.4×10^7 liters of water per 1000 MWe-year is expected to evaporate to the environment from such ponds. The radioactive components of the raffinate, ^{226}Ra , ^{230}Th , and uranium isotopes, are in nonvolatile form, and are therefore expected to accumulate in the pond.

*Corresponds to $\sim 3 \times 10^{-6}$ μCi ^{226}Ra /liter and 3×10^{-3} μCi natural uranium/liter. (7)

Solid wastes from the hydrofluor process are a nonvolatile ash comprising iron, calcium, magnesium, copper, and other nonvolatile fluorides. The quantity of ash produced by hydrofluorination is about 40 MT/1000 MWe-year.⁽¹⁾ The ash is continually removed from the process and reacted with fresh fluorine to recover the uranium, thus minimizing uranium concentrations in the wastes. The final ash residue, containing traces of thorium and protactinium, is packaged and shipped to a licensed commercial waste burial site. The total activity involved is about 0.28 Ci/1000 MWe-year. No similar volume of solid ash waste is produced by wet solvent extraction fluorination.

Commercial plants using each process, hydrofluor and wet solvent extraction, have been built and operated with capacity to process at least 5000 MTU/year, at a single plant, into uranium hexafluoride. To meet domestic requirements, the industry currently produces about 20,000 MT⁽⁸⁾ of uranium as hexafluoride annually.

Uranium hexafluoride prepared by the processes described above is used in isotopic enrichment operations, performed to increase the concentration of the fissile ^{235}U isotope in the uranium. Isotopic enrichment of uranium is necessary to provide fuel for a light-water moderated nuclear reactor. The concentration of ^{235}U in natural uranium is about 0.7% and the required content for the current generation of reactors is 2 to 4%. About 116,000 kilograms (kg) of separative work units (SWU) are required to prepare enough uranium for the production of 1000 MWe-year of power. A separative work unit is a measure, expressed in mass units, for example, kilograms, of the effort expended to separate a quantity of uranium of a given assay into two components, one having a higher percentage of ^{235}U and one (tails) having a lower percentage.

Gaseous diffusion is the technology developed in the U. S. for performing the enrichment operation. The gaseous diffusion process is based on the principle that the rate at which a gas moves through a small hole is related to the speed, and therefore to the masses, of the gas molecules. The enrichment of uranium is accomplished by applying this principle to use of a porous barrier through which the gas molecules diffuse. This effect is multiplied by using a large number of successive barriers, or stages, called a cascade.

In addition to the enriched UF_6 product, the gaseous diffusion plants produce a UF_6 stream depleted in ^{235}U , called "tails." The selection of a tails assay, that is, the ^{235}U content, determines how an enrichment plant will be operated, the separative work involved, and, ultimately, the amounts of ore that will be mined to sustain a given power production schedule. The depleted uranium in the "tails" of the enrichment operation may have a ^{235}U content of 0.2 to 0.3%. This material is stored for future uses at the enrichment plants as solid UF_6 in cylinders or as UF_4 in drums.

2.7

The radioactive component of gaseous wastes from the enrichment operation consists of airborne uranium, estimated at about 0.7 kg/1000 MWe-year,⁽¹⁾ and does not produce needs for shipping or storage under current standards.

Liquid wastes from enrichment that contain radioactivity (uranium) include liquid wastes from process cleanup operations (containing fluorides, nitrates, and iron) and liquid wastes from auxiliary production facilities (containing fluorides, nitrates, chlorides, iron, and ammonia). The quantity of uranium in liquid effluents supporting the production of 1000 MWe-year of power has been estimated at about 10 kg.⁽¹⁾

Solid wastes generated during enrichment come from liquid effluents from process cleanup operations, and are collected in holding ponds. The sludge collected in this manner contains soil runoff from groundwater, small quantities of precipitated metals, various suspended solids, and a detectable amount of radioactivity. The amount of solids attributable to the plant operations is less than 1 MT/1000 MWe-year. This is classified as a nonhigh-level, nontransuranic solid waste and is disposed of by burial onsite. Metallic equipment and scrap are also generated.

2.1.4 Fuel Fabrication (1,3,9-11)

In a fuel fabrication plant, the following main operations might be carried out:

- . UF_6 is converted to UO_2 .
- . Pellets of the UO_2 are formed and sintered to the desired density.
- . The pellets are loaded into Zircaloy tubes.
- . The tubes are fitted with end caps which are then welded into place.
- . The tubes (fuel elements) are assembled into fixed arrays to be handled as fuel assemblies.

An additional operation consists of recovery of uranium from scrap and material failing to meet specifications. The special characteristics of mixed oxide (UO_2 - PuO_2) fuel fabrication plants are described in Section 2.5.

Ten commercial plants licensed by the AEC (now the Nuclear Regulatory Commission, NRC) are presently capable of performing all or part of the fuel fabrication operations. Three of the plants use enriched UF_6 as feed material and produce completed light water reactor fuel assemblies. Four other plants use enriched UF_6 as feed material for producing UO_2 powder or UO_2 pellets. The other three plants use UO_2 powder or UO_2 pellets as feed material and produce completed fuel assemblies.

Assuming the feed material for the plant is enriched UF_6 , received in cylinders from the gaseous diffusion plant, the process steps involved in the currently predominant ammonium diuranate (ADU) process are as follows:

1. UF_6 is heated and transferred as a gas to a hydrolysis reactor.
2. Hydrolysis results from the interaction of the gaseous UF_6 with water forming a UO_2F_2 solution.
3. Precipitation of ADU is accomplished by adding ammonium hydroxide.
4. The ADU slurries are concentrated by centrifuging or filtering.
5. The ADU is dried by heating.
6. The dried ADU is calcined to UO_2 powder in a reducing atmosphere (hydrogen and nitrogen).

Mechanical operations involve the following major steps:

1. pretreatment of UO_2 powder by comminution, compaction, and granulation to desired consistency
2. pelletizing
3. sintering of the pellets
4. grinding to finished dimensions
5. washing and cleaning the pellets
6. loading the pellets into metal tubes and welding the end caps
7. assembling finished fuel elements to form complete fuel assemblies.

Much of the off-specification material, revealed by the frequent testing and inspection procedures, can be recycled into the fabrication line. However, some of this material must be processed through a scrap recovery cycle involving the following principal steps:

1. dissolution of the uranium in nitric acid, forming uranyl nitrate
2. purification of the uranium through solvent extraction
3. precipitation of a hydrated uranium oxide with ammonium hydroxide and hydrogen peroxide
4. conversion to U_3O_8 by drying and
5. reduction to UO_2 powder.

The recovered uranium is returned to the main powder-pellet production system after step 4 or 5 of the scrap recovery cycle.

Cleanup of both liquid and gaseous waste streams generates some solid wastes requiring management operations. Process off-gases from the UF_6 conversion and scrap recovery dissolution operations are passed through scrubber solutions, demisters, and then through a high-efficiency particulate air (HEPA) filter before release to the atmosphere. Exhaust air from enclosures, equipment, and areas where UO_2 powder is handled is drawn through HEPA filters before release. Off-gases from incineration of miscellaneous combustible wastes are treated by a scrubber-demister or a bag filter followed by a HEPA filter.

Although the thorium decay products of uranium are separated from the uranium product at the milling plant, the ^{234}Th decay product of ^{238}U , with a half-life of 24 days, grows back. The activity released from liquid raffinate

holding ponds is estimated at about 20 mCi/1000 MWe-year from uranium, 10 mCi/1000 MWe-year from ^{234}Th , and 10 mCi/1000 MWe-year from ^{234}Pa .

Solid radioactive wastes of the fabrication plant consist mainly of contaminated CaF_2 . Water from the effluent air scrubber systems contains most of the fluoride released during operation. The combined liquid effluent from the UF_6 to UO_2 conversion process and the scrap recovery operations is treated with lime to form a calcium fluoride precipitate. The CaF_2 is filtered from the waste stream, and the water is released to a holding pond or lagoon. The calcium fluoride precipitate filtered from the liquid waste stream is buried in onsite trenches that are then backfilled. The total quantity produced is estimated at about 26 MT/1000 MWe-year. The 26 MT has a volume of about 8.4 m^3 . The CaF_2 solids buried onsite contain about 0.06 Ci of uranium/1000 MWe-year.

2.1.5 Irradiation In Reactor^(12,13)

At the reactor, fresh-fuel assemblies may be temporarily stored, inspected, and tested prior to use. The assemblies are placed in the reactor core, and each is burned at a specific power and for a total burnup* that may be the result of a complex decision process aimed at overall economic optimization. The decision process may include factors such as fraction of core to be replaced (typically 1/5 to 1/3 annually), fuel composition and design, uranium enrichment, physical relocation of assemblies, and length of subsequent operating cycles.⁽¹²⁾ Upon removal from the core, an assembly may be stored outside the core region for 10 days, then removed to the reactor fuel storage pool and stored for 120 to 180 days before shipping. Wastes formed at the reactor include solid byproducts of the cleanup of reactor liquid streams and miscellaneous solids, described in Section 2.2, and also spent fuel, described in Section 2.3. Discarded reactor components constitute a waste stream also described in Section 2.2.

2.1.6 Reprocessing of Irradiated Fuel

Reprocessing of irradiated fuel recovers usable fissile material from spent reactor fuel elements. During the operation of a reactor, the buildup of fission products and depletion of fissile material requires that the operator replace between one-fifth and one-third of the fuel assemblies annually and redistribute the other partially spent fuel in the reactor core. The

*Specific power, commonly expressed in units of thermal megawatts per metric ton of uranium (MW/MTU), represents the rate at which energy is extracted from the fuel, and the burnup, commonly expressed in thermal megawatt-days per metric ton of uranium (Mwd/MTU), represents the total energy extracted. For MOX fuel, the unit of fuel is a metric ton of heavy metal (MTHM), i.e., a metric ton of (U + Pu).

discharged fuel elements contain about one-third of the ^{235}U that was in the fuel prior to irradiation, and the fissile plutonium that was produced from ^{238}U during irradiation. The recovered uranium can be recycled in reactor-reload fuel after addition of ^{235}U to replace the spent fraction, and the plutonium can be stored or used in LWR reload fuel or as fuel in fast breeder reactors. Details of the reprocessing operation are described in Section 2.4 and projections of the reprocessing load and fuel processed are given in Section 3.

2.1.7 Transportation

The flow of materials between facilities of the nuclear fuel cycle, diagrammed in Figure 2.1, briefly indicates the various types of materials-transport situations that may arise in a complete LWR fuel cycle. These various transportation activities are detailed in Section 22. Also, projections of shipments are given in Section 3.2, including fresh fuel shipments by truck to reactors, shipments of plutonium by truck to fabrication plants, spent fuel shipments to reprocessing plants, and shipment of 10-year-old high-level waste and transuranic (TRU) contaminated material in containers that may have held these wastes from 5 to 10 years.

2.1.8 Alternative Fuel Cycles

The LWR fuel cycle shown previously in Figure 2.1 can be changed in several ways, resulting in significant differences in waste management requirements. Spent fuel from the reactor contains a modest concentration of fissile isotopes (^{235}U , ^{239}Pu , and ^{241}Pu) and a significant concentration of fertile ^{235}U material (^{238}U). The spent fuel may, as shown in Figure 2.1, be transported to a reprocessing plant at which the fissile values are extracted and purified for reuse. Alternatively, spent fuel can be treated as waste and transported directly to a waste storage or a federal repository (i.e., "throw-away" fuel cycle). Further, plutonium that has been isolated by reprocessing of spent LWR fuel can be fabricated into mixed oxide (MOX) fuel for LWR, stored for other uses (e.g., fast reactor fuel), or treated as waste. The alternative fuel cycles are shown in Figure 2.2.

The impact of the alternatives on waste management is shown in Figure 2.3. The impact on reactor wastes of the alternative fuel cycles is minimal and will not be detailed. The composition of the spent fuel, however, is influenced by the fuel cycle mode. Because the "throwaway" fuel cycle does not include reprocessing, ~200-times the amount of uranium and plutonium normally expected in the high-level waste from the reprocessing plant will be discarded in the spent fuel assembly. On the other hand, the lack of plutonium fuel in this fuel cycle option reduces significantly the transuranic* (TRU) content of the

*Transuranics are, by proposed regulation,⁽¹⁴⁾ all elements with atomic numbers greater than 92, for example, Np, Pu, Am, Cm.

FIGURE 2.2. Alternative Fuel Cycles

	REACTOR WASTES	HIGH- LEVEL WASTE	REPROCESSING WASTE	MOX PLANT WASTE
OPTION				
"THROWAWAY"	X	X ^(a)		
URANIUM RECYCLE	X	X	X ^(b)	
U-Pu RECYCLE (REFERENCE)	X	X	X	X

a. As intact fuel assemblies.
 b. Includes plutonium.

FIGURE 2.3. Waste Sources from Alternative Fuel Cycles

spent fuel. The alternative of reprocessing spent fuel to recover the unused ^{235}U and isolation of plutonium for storage (rather than recycle in MOX elements) is a fuel cycle alternative that generates high-level waste and other reprocessing plant wastes but no waste from MOX fabrication. The high-level waste produced in this alternative is similar to the waste formed by processing of mixtures of UO_2 and MOX fuel except that there is a lower trans-uranium content and a corresponding reduction in the neutron emission rate. Additional descriptions of fuel cycle alternatives are given in Sections 2.3 and 2.4.

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2.2 WASTES FROM REACTORS

Based on the state of technological development and of commercial construction activity and commercial intentions, light water reactors of the boiling water reactor (BWR) type and pressurized water reactor (PWR) type are expected to provide a major portion of all nuclear power produced in the U. S. through at least the year 2000. Most of the radioactive wastes generated in power production by light water reactors will remain in the spent fuel, and be collected not at the reactor site, but at the fuel reprocessing facility when the spent fuel is disintegrated by chopping and leaching processes. A minor portion, however, will normally be present for potential release at the reactor site. This radioactivity results from neutron activation of structural materials near the core, corrosion products and chemical additives, fission of tramp uranium on the surfaces of fuel cladding, and failure of a small fraction (0.25%) of fuel cladding that results in release of constituents* of irradiated fuel into the primary coolant system. All soluble radioactivity that becomes available for potential release at the LWR site is originally introduced into either the primary coolant or the fuel storage pool coolant. Minor portions of these liquids spread to various parts of the LWR plant by leakages, and therefore radioactivity appears in various waste streams of the LWR plant.

The present discussion is limited to the liquid waste streams. Operations to remove radioactivity and both dissolved and suspended solids are applied to all these liquid streams before either recycle for use at the plant or discharge. The byproducts of the operations on liquid streams, therefore, will contain most of the radioactivity from the LWR site requiring waste management operations such as shipment and storage. These byproducts are in the form of wet solids. Also, some of the radioactivity at the LWR site will be incorporated in dry wastes of miscellaneous form and composition.^(1,2)

Some radioactivity from the reactor site, including noble gases, tritium, and minor amounts of other isotopes, is dispersed to the environment in amounts and concentrations limited by current regulations. These wastes represent no demand at this time for further waste management operations and are therefore not considered here. The general flow patterns of radioactive wastes in a boiling water reactor (BWR) and in a pressurized water reactor (PWR) are diagrammed in Figures 2.4 and 2.5.

In summary, in addition to the accumulation of miscellaneous dry solid wastes, an important determinant of the origin and nature of wastes emerging from reactors is the management of internal waste streams, the cleanup

*Mainly gaseous and volatile constituents; a minute fraction of the oxide fuel is water-soluble, and the penetration of water into a failed element is unlikely except during reactor shutdowns.

FIGURE 2.4. Typical Flow Pattern of Radioactive Wastes in the BWR System

FIGURE 2.5. Typical Flow Pattern of Radioactive Wastes in the PWR System

operations applied to them, and the wet solids resulting from these operations. Therefore, the management of internal liquid streams and the cleanup operations applied to them are described below.

2.2.1 Management and Cleanup Operations on Internal LWR Liquid Streams

The processes applied to liquid reactor streams and producing wet solid wastes are filtering, demineralizing (ion exchange), evaporating, and (less commonly) centrifuging and reverse osmosis.⁽¹⁻⁵⁾ Prior to these processes, the various streams in a light water reactor are classified and segregated or collected according to their radioactivity and dissolved solids content in order to combine waste streams having similar characteristics and requiring similar treatment. Streams of low dissolved solids content ("clean" wastes) can be purified directly by demineralization. On the other hand, streams of high dissolved solids content ("dirty" wastes) are unsuitable for demineralization; they are treated by evaporation, producing a bottoms slurry that is concentrated in solids, and a purified, overhead stream that can be purified further by demineralization. Reverse osmosis has been suggested for use in reducing the dissolved solids content of laundry wastes before discharge.⁽¹⁾ The reject stream from reverse osmosis would be concentrated in both dissolved and suspended solids. Centrifugation, to dewater some of these wet solids, has been used less than the simpler process of settling and decantation, but will probably be used more widely in the future. An outline of the general management scheme is shown in Figure 2.6. These operations are detailed in Section 10.

2.2.2 Management of Primary Coolant

Radioactivity enters the wet wastes through the direct processing of primary coolant or of other streams containing primary coolant because of various transport processes and leakage. For the wet wastes, therefore, the radioactivity level and the care in management will increase generally in proportion to the concentrations of primary coolant that they normally contact. The origin of some of the characteristics of these wastes can be inferred from the following general description of primary coolant management in BWRs and PWRs.

In a BWR, primary coolant water is converted to steam by heat from the fuel elements in the core. The steam expands through a turbine and then is condensed. The condensed steam is processed through the condensate cleanup system, collected, and returned to the reactor for reuse. A separate side stream of primary coolant is continuously withdrawn and processed through the reactor cleanup system (demineralizers) and returned to the reactor vessel. Both cleanup systems removed suspended solids and ionic impurities from the primary coolant. The materials collected by the demineralizers are removed periodically by chemical regeneration or by replacement of resins. The liquid

FIGURE 2.6. General Scheme for Management of Liquid Wastes at Reactors

wastes are processed in the liquid waste treatment system, and the ion exchange resins are transferred to the solid waste treatment system and prepared for offsite shipment.

Liquid leakage occurring through valve stems, pump seals, and flanged connections is collected in drains and routed to the appropriate liquid waste treatment systems. Additional sources of liquid wastes are: the liquid streams needed to sluice (transfer), backwash, regenerate, and rinse demineralizer resins; wastewater from the laundry, personnel showers, laboratory drains, and decontamination areas; and water collected in equipment drains and floor drains.

In a PWR, primary coolant water circulates through the reactor core, where it removes the heat from the fuel elements. The pressurized primary coolant water flows from the core into the steam generator, where heat is transferred from it to the secondary coolant water, forming steam. The steam expands through the turbine and is then condensed and returned to the steam generators. The primary coolant water flows back to the reactor core.

The primary coolant is continuously purified by passing a side stream through filters and demineralizers in the chemical and volume control system. The purity of the primary coolant must be maintained to reduce corrosion, to prevent fouling of heat surfaces, and to keep releases to the environment to a low level.

Chemicals are added to the primary coolant to inhibit corrosion and/or improve fuel economy. Lithium hydroxide is added for pH control to reduce corrosion. Boron is also added to the primary coolant as a neutron absorber (shim control). At a later stage, boron is removed from the primary coolant, if adjustment of concentration is necessary, through the chemical and volume control loop (shim bleed). The shim bleed is processed through an evaporator, and the boron in the evaporator bottoms is either reused or packaged as solid waste. The evaporator distillate may be recycled to the reactor coolant system as makeup water or discharged to the environment.

By leakage through valve stems and pump shaft seals, some primary coolant escapes into the containment and into the auxiliary buildings. A portion of the leakage evaporates, and the remainder is collected in drains to be processed through the liquid waste treatment systems.

Leakage of primary coolant into the secondary coolant in the steam generator is the only source of radioactivity in the secondary coolant system. Water or steam leakage from the secondary system adds significant amounts to the liquid and gaseous radwaste treatment systems. In a secondary system maintained by blowdown,* the radioactivity introduced into that system will

*Blowdown is the removal of a portion of the recirculating cooling water containing concentrated dissolved salts to prevent their buildup in the system.

eventually appear in solids used to treat streams that collect leaks from the secondary system. However, in a secondary system maintained by demineralization, the radioactivity introduced to that system will eventually be borne on the resins of that demineralizer.

2.2.3 Typical Waste Management Schemes and Solid Waste Generation Rates

Flow sheets for liquid waste management at LWRs are not standardized. Considering waste generation rates, it is convenient to classify light water reactors into four main types:⁽⁷⁾ BWRs and PWRs that treat the condensate with deep bed demineralizers and BWRs and PWRs that treat the condensate with powdered-resin filter-demineralizers. Figures 2.7 through 2.10 illustrate simplified schemes for liquid waste management and solid waste generation rates to be expected for these four types. The activity levels are based on an assumed cladding failure of 0.25%.*

2.2.4 Summary of Reactor Waste Volumes

The reactor waste volumes listed in Figures 2.7 through 2.10 are summed according to primary waste type and listed in Table 2.1.

2.2.5 Reactor Component Wastes

Replacement of and maintenance on reactor components (e.g., control rod guides, instruments, control rods, support plates, etc.) produce waste that is radioactive because of activation products formed during service within the high neutron flux region of the reactor. The activation products are an inherent part of the solid metal structures in which they were generated and thus become available contamination only upon deterioration of the very highly corrosion-resistant metals. The metallic components are usually sufficiently radioactive to require shielding for disposal and may need to be cut or compacted to fit into modest sized containers. Burial in commercial facilities is the probable disposition since this waste constitutes a non-TRU, intermediate level β - γ waste with little or no mobile, radioactive contamination. Data on quantities or properties estimated for a 1000 MWe reactor are not available.

*This cladding failure percentage indicates that 0.25% of the total thermal power of the reactor is produced in fuel rods that have developed cladding defects. The percentage of total fission products, contained in irradiated fuel, that escape into the primary coolant would be several orders of magnitude lower.

FIGURE 2.7. Sources and Characteristics of Non-Gaseous Wastes from 1-GWe BWR with Deep-Bed Condensate Cleanup

FIGURE 2.8. Sources and Characteristics of Non-Gaseous Wastes from 1-GWe BWR with Powdered Resin Condensate Cleanup

FIGURE 2.9. Sources and Characteristics of Non-Gaseous Wastes from 1-GWe PWR with Powdered Resin Condensate Cleanup

FIGURE 2.10. Sources and Characteristics of Non-Gaseous Wastes from 1 GWe PWR with Deep-Bed Condensate Cleanup

TABLE 2.1. Summary of Non-Gaseous Reactor Waste Volumes^(a)

Primary Waste Type	Amounts in ³ m/1000 MWe-year			
	BWR	BWR	PWR	PWR
	Deep-Bed ^(b)	Powdered Resin ^(c)	Deep-Bed	Powdered Resin
Combustible β - γ	351 ^(d)	392 ^(e)	223 ^(f)	301 ^(f)
Non-Combustible Solid β - γ ^(g)	57	57	46	46
Non-Combustible Liquid β - γ ^(h)	280	0.14	317	23

a. Derived from Reference 7; all volumes on uncompacted basis.

b. Deep-bed demineralizer used for condensate cleanup.

c. Powdered resin filter-demineralizer used for condensate cleanup.

d. Resins + aqueous filter sludges (assumes combustible precoats) + 80% of miscellaneous dry wastes.

e. Resins + aqueous filter cartridges + filter sludges + 80% of miscellaneous dry wastes.

f. Resins + aqueous filter cartridges + 80% of miscellaneous dry wastes.

g. 20% of miscellaneous dry wastes.

h. Evaporator slurries.

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2.3 WASTES FROM SPENT FUEL STORAGE FACILITIES

A spent fuel storage facility (pool) offers a number of services, including storage, cooling (usually by transfer of heat to the environment through a secondary cooling system and heat exchanger, e.g., an evaporative cooling tower), shielding, inspection, and control of radioactive materials released from both intact and failed fuel.^(1,2) Pools will be a source of both wet and dry solid wastes that will require shipping, storage, and associated management operations. These wastes consist mainly of the byproducts of operations used to remove radioactive materials introduced into the pool water by both intact and failed fuel.

Pools are presently located at Federal sites including the Savannah River Plant, the Idaho National Engineering Laboratory, and the Hanford Plant. These pools are associated with both government-owned reactors and fuel reprocessing plants. Other pools are located at nuclear power stations and commercial fuel reprocessing plants. Although commercial plants are not reprocessing fuel at present, the fuel pools at both the Nuclear Fuel Services (NFS) Plant* and Midwest Fuel Reprocessing Plant (MFRP)** are already used for storage of spent fuel assemblies, and the pool at Barnwell Nuclear Fuels Plant (BNFP)[†] is expected to be put into service before reprocessing begins.⁽³⁾ In addition to the above locations, independent pools, not associated with any other nuclear facility, have been proposed to alleviate the lack of storage space for fuel presently being discharged from operating nuclear power stations, but for which storage capacity is rapidly diminishing. Several design studies are under way in an attempt to substantially increase the capacity of existing water basins.

The following sections discuss the characteristics of spent fuel, the operations of fuel pools, and finally the wastes emerging from these operations.

2.3.1 Description of Spent Fuel

Light-water reactor fuel assemblies^{††} are composite units made of fuel elements^{††} in a geometric cluster and held together by end pieces and a number of element spacers. BWR and PWR fuel assemblies differ significantly, but the basic component of each is the fuel element. A fuel element of either type is a long section of metal tubing filled with ceramic pellets of uranium dioxide

*West Valley, NY.

**General Electric Co., Morris, IL.

[†]Allied-General Nuclear Services, Barnwell, SC.

^{††}"Fuel Assembly: A grouping of fuel elements which is not taken apart during the charging and discharging of a reactor core--Fuel Element, The smallest structurally discrete part of a reactor or fuel assembly which has nuclear fuel as its principal constituent."⁽⁷⁾

or mixed uranium-plutonium dioxide and closed at both ends (see Figure 2.11). The BWR⁽⁴⁾ cladding tube is typically larger (~1.4-cm diam) and the finished element longer (~400 cm) than the PWR element. A typical fuel loading is ~3 kg of heavy metal in the oxide form per BWR fuel element. A PWR fuel element,⁽⁵⁾ on the other hand, has cladding with a 1-cm diam and a 380-cm length. A typical PWR fuel loading is ~2 kg of heavy metal per element.

Differences between completed BWR and PWR fuel assemblies are even more noticeable. In the BWR assembly (Figure 2.12), the fuel elements are arranged in a 7- x 7-element square array, but advanced fuels are to contain 8 x 8 arrays of 1.25-cm diam elements.^(4,6) The total fuel-loading is ~200 kg of heavy metal for an average 64-element BWR fuel assembly. The individual elements are secured by upper and lower tie-plates and spacer-grids at intervals along the element cluster. The entire assembly is covered by a thin Zircaloy sheet-metal box called a fuel channel. The fuel channel is normally reused following withdrawal of an assembly. The total weight of a completed BWR fuel assembly is typically ~280 kg (excluding fuel channel). It has a 14- x 14-cm cross section and is 4.5 m long.

The PWR assembly (Figure 2.13) contains 264 fuel elements in a square (17 x 17) array of locations, some of which are occupied by reactivity-control rods,* instrumentation,* and burnable-poison elements.^(5,9) An average fuel loading for a complete PWR assembly is ~460 kg of heavy metal. A PWR assembly has no cover or coolant channel, being secured by top and bottom nozzle fixtures and spring-clip grids spaced along the length of the element cluster. A completed PWR fuel assembly weighs ~670 kg, has a cross section of 22 x 22 cm, and is about 4 m long.

2.3.2 Properties of Spent Fuel

For purposes of describing compositions of waste streams, the physical differences in BWR and PWR fuel assemblies and the imbalance of numbers of the two types of reactors in the industry are synthesized into a hypothetical "reference LWR" fuel-assembly. Table 2.2 characterizes the reference assembly. Table 2.3 details the chemical composition. The oxide fuel would be contained in a square array with 2.57 kg of heavy metal per element. Total weight of this reference assembly is set at 454 kg. The reference LWR fuel assembly contains fuel-cladding tubes that are made of Zircaloy with Zircaloy end plugs. Top and bottom assembly tie-plates or nozzle fixtures will be fabricated from type 304 stainless steel. Element-spacing fixtures will be of Inconel. The fuel pellets are UO_2 or $(\text{U}, \text{Pu})\text{O}_2$.

*Control rods and in-core instrumentation are not part of a PWR assembly and are removed from the array prior to reactor discharge.

FIGURE 2.11. Typical PWR Fuel Element

FIGURE 2.12. BWR Fuel Assembly⁽⁴⁾

FIGURE 2.13. PWR Fuel Assembly⁽⁵⁾

TABLE 2.2. Characteristics of Unirradiated LWR Fuel Assemblies

	PWR ⁽⁵⁾	BWR ⁽⁴⁾	"Reference" LWR ^(g)
Overall assembly length, m	4.059	4.470	-----
Cross section, cm	21.4 x 21.4	13.9 x 13.9	-----
Fuel element length, m	3.851	4.064	-----
Active fuel height, m	3.658	3.759	3.70 ^(e)
Fuel element OD, cm	0.950	1.252	1.16 ^(e)
Fuel element array	17 x 17	8 x 8	121 ^(e)
Assembly total weight, kg	668.6	279	454.32
Uranium/Assembly, kg	461.4	188.7	311.4 ^(h)
MO ₂ /Assembly, kg	523.4	214.1	353.28
Zircaloy/Assembly, kg	129.7 ^(a)	56.7 ^(b)	89.55
Hardware/Assembly, kg	15.5 ^(c)	8.2 ^(d)	11.485 ^(f)
Total metal/Assembly, kg	145.2	64.9	101.03

a. Includes Zircaloy control-rod guide thimbles.

b. Includes Zircaloy fuel-element spacers.

c. Includes 10 kg SS nozzles, 5.5 kg Inconel-718 grids.

d. Includes SS tie-plates, negligible amount of Inconel springs.

e. Specified only to estimate fuel content of each element of 2.57 kg heavy metal.

f. Includes 9 kg SS end pieces, 2.5 kg Inconel-718 spacers.

g. Based on discharges of fuel from PWR and BWR reactors with installed capacity ratio of 2 PWR:1 BWR, using the same burn-up and fuel assemblies of advanced GE and Westinghouse design (Reference 4,5). The reference fuel assembly produces a metallic stream comparable to processing of a mix of PWR and BWR fuel.

h. kg of heavy metal (U + Pu).

TABLE 2.3. Chemical Composition of the
Reference LWR Fuel Assembly
(As Fabricated)

Values in kg/Assembly (kg/MTHM) ^(a)		
Heavy Metal Oxide Fuel ^(b)		353.3 (1,134)
PuO ₂	5.27 (16.9)	
UO ₂	348.03 (1,120)	
Zircaloy-4		89.55 (287.5)
Sn	1.43 (4.59)	
Fe	0.202 (.648)	
Cr	0.112 (.360)	
Zr	87.85 (282)	
Stainless Steel Type 304		9.01 (28.9)
Cr	1.71 (5.49)	
Ni	0.811 (2.60)	
Mn	0.180 (.578)	
Si	0.090 (.289)	
P	0.018 (.0578)	
C	0.014 (.0449)	
Se	0.014 (.0449)	
S	0.005 (.0161)	
Fe	6.16 (19.8)	
Inconel-718		2.475 (7.96)
Ni	1.31 (4.21)	
Cr	0.470 (1.51)	
Fe	0.446 (1.43)	
Nb/Ta	0.124 (.398)	
Mo	0.074 (.238)	
Co	0.020 (.0642)	
Ti	0.020 (.0642)	
Al	0.015 (.0482)	
C	0.001 (.00321)	

a. MTHM = metric ton of heavy metal (U, Pu).

b. Data calculated from fuel charge amounts in
Reference 8.

After irradiation at 35 MW/MTHM specific power and 25,000 MWd/MTHM burnup, in the core of a LWR, a spent-fuel assembly will have a total thermal power of 640 kilowatts (kW). After 120 days cooling the heat generation rate has decayed to 8.4 kW per assembly. These data are shown in Table 2.4. A detailed list of fission products, actinide elements, and metallic construction elements, together with the anticipated radiations for various cooling times is also shown in Table 2.4. The spent-fuel assemblies will have a beta-gamma radiation dose rate at discharge of $>10^6$ Roentgen/hour (R/hr) at the most radioactive surface.

For the purposes of this document, it is assumed that the LWR spent-fuel assemblies contain no leaking and/or ruptured fuel elements. Thus, it is assumed that, except for emission of heat and radioactivity, the assemblies are physically unaltered from their fabricated conditions. It is recognized that a modest fraction of assemblies will contain leaking elements and that these require special handling. This impact on the overall waste management processes described in these sections is assumed to be small.

2.3.3 Quantities of Spent Fuel

It is assumed that a LWR will generate 45.6 MTHM of spent fuel in the production of 1000 MWe-year of power.* Based on the reference LWR fuel assembly containing 311.4 kg of heavy metal, 147 spent-fuel assemblies will be discharged for each 1000 MWe-year. A 1500-MTHM/year fuel reprocessing plant will process about 4800 of the reference assemblies each year, or 16 assemblies per day for 300 days. This rate is equal to fuel output from 33,000 MWe-year power generation by LWRs. Table 2.5 gives the weight distributions of the reference LWR assembly at various appearance rates.

2.3.4 Characteristics and Operations of Reactor Pools

A typical $\sim$ 1000 MWe reactor pool can store as much as 1 2/3 reactor cores, i.e., about 100 to 250 MTHM of fuel.** During storage, much of the activity associated with short-lived isotopes will decay. The presence of short-lived activity distinguishes reactor pools from the pools both at fuel reprocessing plants and independent locations. At some reactors, assemblies are transferred from core positions to storage in the pool via a canal that joins the core and the pool. Thus, in this transfer, primary coolant mixes with pool water and the pool water can become contaminated. Radioactivity can be introduced into a reactor pool from mixing with primary coolant, from leaching of fuel from assemblies with failed cladding, and from shedding of deposits from the surfaces of assemblies.

*Based on 25,000 MWd/MTHM burnup as currently apparent in spent fuel and an efficiency of 0.32 MWe/MWth.

**PWR core, 89 MTHM; BWR core, 138 MTHM. (4,5)

TABLE 2.4. Chemical, Radiochemical, and Thermal Properties of
Irradiated Reference LWR Assembly at Three Cooling
Times (a,b)

Element	Grams/ Assembly ^(c)	Curies/Spent-Fuel Assembly		
		120-days	1-year	10-year
<u>Fission products^(d)</u>				
H ^(e)	4.2 E-2	1.7 E2	1.6 E2	9.7 E1
Ge	9.0 E-2	---	---	---
As	3.0 E-2	---	---	---
Se	1.3 E1	9.0 E-2	9.0 E-2	9.0 E-2
Br	5.3 E0	---	---	---
Kr	7.8 E1	2.1 E3	2.0 E3	1.1 E3
Rb	7.5 E1	1.6 E0	1.8 E-4	4.7 E-6
Sr	1.9 E2	7.0 E4	1.9 E4	1.3 E4
Y	1.0 E2	1.0 E5	2.1 E4	1.3 E4
Zr	8.1 E2	1.4 E5	1.0 E4	6.0 E-1
Nb	5.6 E-1	2.5 E5	2.2 E4	2.9 E-2
Mo	8.1 E2	---	---	---
Tc	2.0 E2	3.4 E0	3.4 E0	3.4 E0
Ru	5.6 E2	2.0 E5	8.7 E4	1.8 E1
Rh	1.3 E1	2.0 E5	8.7 E4	1.8 E1
Pd	3.4 E2	3.1 E-2	3.1 E-2	3.1 E-2
Ag	2.1 E1	1.8 E4	9.3 E3	1.1 E0
Cd	1.9 E1	6.5 E1	9.3 E0	5.0 E-1
In	4.0 E-1	5.0 E-1	1.5 E-2	---
Sn	1.2 E1	8.0 E2	2.2 E2	1.9 E-1
Sb	4.3 E0	2.6 E3	2.2 E3	2.2 E2
Te	1.2 E2	9.0 E3	1.5 E3	5.6 E1
I	6.2 E1	1.0 E1	8.5 E-3	8.4 E-3
Xe	1.3 E3	6.0 E1	4.0 E-6	---
Cs	6.8 E2	6.8 E4	6.0 E4	2.2 E4
Ba	3.4 E2	2.5 E4	2.4 E4	2.0 E-4
La	2.9 E2	9.0 E2	1.5 E-3	---
Ce	5.9 E2	2.9 E5	1.4 E5	4.7 E1
Pr	2.7 E2	2.6 E5	1.4 E5	4.7 E1
Nd	1.2 E3	1.0 E2	2.0 E-3	---
Pm	3.0 E1	3.4 E4	2.8 E4	2.6 E3
Sm	1.9 E2	3.4 E1	3.4 E1	3.1 E1
Eu	3.7 E1	3.0 E3	2.6 E3	1.2 E3
Gd	1.8 E1	2.7 E0	1.9 E0	---
Tb	5.3 E-1	1.1 E2	7.1 E0	---
Grams/Assembly	<u>6.8 E3</u>	---	---	---
Curies/Assembly		<u>1.7 E6</u>	<u>6.5 E5</u>	<u>7.4 E4</u>
<u>Light elements^(f)</u>				
C	1.7 E1	2.5 E-1	2.5 E-1	2.5 E-1
Al	1.5 E1	---	---	---
Si	9.1 E1	---	---	---
Ti	2.0 E1	---	---	---
Cr	2.3 E3	4.9 E2	1.1 E0	---
Mn	1.8 E2	1.7 E3	9.9 E1	5.4 E-2
Fe	6.9 E3	2.2 E3	1.8 E3	1.6 E2
Co	2.0 E1	2.8 E3	1.6 E3	4.5 E2
Ni	2.1 E3	7.9 E1	7.9 E1	7.4 E1
Zr	8.8 E4	2.8 E3	2.0 E2	2.4 E-2
Nb	1.2 E2	1.8 E3	1.5 E2	4.2 E-3

TABLE 2.4. (contd)

		Curies/Spent-Fuel Assembly		
Element	Grams/ Assembly ^(b)	120-days	1-year	10-year
<u>Light elements^(f)</u>				
Mo	7.6 E1	1.5 E-3	1.5 E-3	3.6 E-3
Tc	1.2 E-1	2.0 E-3	2.0 E-3	2.0 E-3
Sn	1.3 E3	7.4 E3	1.9 E3	7.9 E-2
Sb	8.5 E1	5.8 E3	3.5 E3	3.3 E2
Te	4.9 E0	1.6 E3	1.4 E3	1.4 E2
Grams/Assembly	<u>1.0 E5</u>	---	---	---
Curies/Assembly		<u>2.7 E4</u>	<u>1.1 E4</u>	<u>1.2 E3</u>
<u>Transuranics^(d)</u>				
Np	7.9 E1	4.3 E1	4.3 E1	4.3 E1
Pu	5.0 E3	8.2 E4	7.9 E4	5.2 E4
Am	2.9 E2	2.3 E2	3.2 E2	1.2 E3
Cm	1.1 E2	3.1 E4	1.6 E4	5.5 E3
Grams/Assembly	<u>5.5 E3</u>	---	---	---
Curies/Assembly		<u>1.1 E5</u>	<u>9.5 E4</u>	<u>5.9 E4</u>
Total grams/Assembly ^(g)	<u>1.1 E5</u>	---	---	---
Total curies/Assembly		<u>1.8 E6</u>	<u>7.6 E5</u>	<u>1.3 E5</u>
<u>Thermal watts/Assembly</u>				
Fission products		7.1 E3	2.9 E3	2.2 E2
Light elements		1.5 E2	5.9 E1	8.7 E0
Transuranics		<u>1.2 E3</u>	<u>6.6 E2</u>	<u>3.1 E2</u>
Total		<u>8.4 E3</u>	<u>3.6 E3</u>	<u>5.4 E2</u>
<u>Total neutrons</u>				
n/sec-Assembly		1.3 E9	1.2 E9	8.0 E8

- a. Based on recycle core loading of 70% UO₂ fuel and 30% MOX fuel, 25,000 MWd/MTHM burnup and 35 MW/MTHM specific power.
- b. For 8.4 E3 read 8.4×10^3 or 8400; for 4.0 E-1 read 4.0×10^{-1} or 0.40.
- c. Load at 1-year cooling.
- d. Contained in oxide fuel.
- e. As tritium assuming 170.3 Ci/assembly at discharge.
- f. Contained in metallic assembly components.
- g. Excludes 3.38 E5 g UO₂ and 6.7 E2 g oxygen from PuO₂. See Table 2.2 for mass summaries.

TABLE 2.5. Reference LWR Fuel Assembly Weight Distributions at Various Appearance Rates

	All Weights in Metric Tons Per			
	Assembly	1 MTHM	GWe-year	1500 MTHM
Assemblies, No.	-1-	-3.21-	-147-	-4817-
U-Pu weight	0.3114	1	45.6	1500
Oxide fuel weight	0.3533	1.13	51.9	1702
Zircaloy weight	0.0896	0.288	13.2	432
Hardware weight	0.0115	0.037	1.69	55.4
Total weight	0.4544	1.46	66.8	2189

2.3.5 Characteristics and Operations of Pools at Fuel Reprocessing Plants and Independent Locations

The capacity of storage pools at reprocessing plants varies from 90 MTHM at the Midwest plant to an estimated 360 MTHM at the Barnwell plant.⁽³⁾ The data for a "typical" 1500 MTHM/year reprocessing plant has been reduced to ~250 MTHM, based on 800 "reference LWR" assemblies.

The general set of operations carried out at fuel reprocessing plants and independent locations that will result in introduction of radioactivity into the pool and ultimately in generation of wastes requiring shipping and storage, are outlined as follows. On receipt, a shipping cask containing spent fuel is inspected and monitored for exterior radioactivity. If the cask is not contaminated, it is washed outside the plant. However, if the cask is contaminated, it will be washed inside the plant and the contaminated liquids routed to the waste treatment system. The cask is then transferred to a decontamination pit where the cask is vented, and the gases are routed through a filtered exhaust system. If the cask coolant* is found to be contaminated, it is pumped to the waste treatment system, and the interior of the cask is flushed. The cask is then transferred to a separate cask-unloading pool where the fuel assemblies are transferred to storage canisters (Figure 2.14). To avoid stresses on assemblies, they are transferred to storage positions by moving canisters rather than individual assemblies. The loaded canisters are placed in racks for storage.

The lid is replaced on the unloaded cask, the cask rinsed with clean water as it is removed from the cask unloading pool, and drained, if it is the dry-cavity type. The cask is then transferred back to the decontamination pit for

*The Allied-General Nuclear Services spent-fuel cask is designed for use without an internal coolant; personal communication, R. Keely (AGNS) to M. Steindler (ANL).

FIGURE 2.14. Fuel Storage Canister

inspection and additional decontamination. Finally the cask is removed and remounted on the transport vehicle and shipped from the plant site.

At pools associated with commercial reprocessing plants, where there is no communication with primary reactor coolants, and short-lived isotopes have decayed before fuel is introduced to the pool, the chance of radioactive contamination is less than at reactor pools. Probably the most important source of contamination of reprocessing plant storage pools is shedding of the deposit of radioactive materials on the surface of fuel elements. These deposits consist of activated corrosion products and the components of failed fuel. The major corrosion products will be ^{58}Co , ^{60}Co , and ^{54}Mn . The major fission products expected are ^{134}Cs , ^{137}Cs , ^{106}Ru - ^{106}Rh , ^{95}Zr - ^{95}Nb , ^{124}Sb , ^{144}Ce , ^{129}I , and ^{131}I . One other source of contamination of these pools is contaminated coolant from shipping casks.

2.3.6 Waste-Generating Operations and Magnitude of Waste Streams

The operations that generate wastes requiring shipping and storage are mainly filtration and ion exchange. These are done to maintain clarity of the water for observing underwater operation and storage conditions, and to keep radiation levels sufficiently low to avoid excessive exposure of operating personnel. These operations are designed to remove dirt, corrosion products, and radioactive material from the pool water. Filtration may be carried out using porous stainless steel elements in stainless steel housings. A typical ion exchange bed will consist of mixed resin, containing both anion and cation resins and an acid and a caustic regeneration system.

The atmosphere of the pool system is conditioned typically by a ventilation and air conditioning system of $2.8 \times 10^3 \text{ m}^3/\text{min}$ capacity. Also, an exhaust system for venting shipping casks and the hoods that are placed over leaking fuel assemblies will have a capacity of $2.8 \text{ m}^3/\text{min}$ (100 cfm) and be equipped with HEPA filters and iodine absorbers.

The wastes that emerge from these cleanup operations are associated with changing and charging pool water filters, ion exchangers, and iodine absorbers, and with waste treatment and packaging. Thus they will consist of wet solids, including spent resins and filter sludges, and miscellaneous dry solids, including HEPA filters, charcoal or silver zeolites (at reactor pools), clothing, plastic, paper, wood, metal, and rubber (Figures 2.15 and 2.16). Estimates of the waste volume generated from fuel storage pools are shown in Table 2.6.

*~1.6 m³/GWe-year; ~4 Ci/m³ mixed activation and fission products.

FIGURE 2.15. Operations and Waste Generation Involving Reactor Fuel Storage Pools

FIGURE 2.16. Operations and Waste Generation Involving Fuel Storage Pools at Reprocessing Plants and Independent Locations

TABLE 2.6. Summary of Solid Wastes Volumes from Spent Fuel Basins(a)

Facility	Resin Volume m ³ /year	Waste Type
1000 MWe Reactor	~1.6 ^(b)	Combustible, β - γ ^(c)
Reprocessing plant ^(d) (1500 MTU/year)	~0.6 ^(e)	Combustible, β - γ ^(c)

- a. Volumes of silver zeolites, ventilation (HEPA) filters, and aqueous filters specifically identified with only the operation of the spent fuel basin are not available.
- b. Weighted average of values listed in Figures 2.7 through 2.10.
- c. Containing mixed fission products, activation products, and actinides. Activity levels in reactor-generated resins may be ~4 Ci/m³.
- d. With a storage pool containing about 800 "reference" LWR assemblies.
- e. Personal communication, November 1975, R. B. Keely (Allied-General Nuclear Services) to L. E. Trevorow (Argonne National Laboratory).

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2.4 WASTES FROM FUEL REPROCESSING PLANTS

Most of the radioactivity in nuclear wastes is generated in processes which disassemble the reactor fuel and separate uranium, plutonium, and the fission products. Plutonium and uranium are segregated to recover their fissile values for reuse in the nuclear fuel cycle. The wastes are isolated to insure their long-term containment.

For the purposes of this document, the reprocessing plant operations start with the receipt of spent fuel at the reprocessing plant and terminate with the formation and packaging for shipment of plutonium dioxide and uranium hexafluoride. Described here, in brief detail, is a generic reference process. The process description emphasizes the behavior and formation of waste streams rather than the behavior and transport of the products. Fuel reprocessing technology for plants in existence or under development is based on the aqueous reprocessing by methods known as the Purex processes. A reference Purex process flow sheet for a 1500-MTHM per year reprocessing plant⁽¹⁻³⁾ is therefore used.

2.4.1 Process Outline and Sources of Waste

The "head end" process steps include 1) the receipt of fuel in shielded casks at the plant, 2) the storage of fuel for sufficient time to allow decay of radioactivity to levels used as criteria for the reprocessing operations, 3) disassembly of fuel assemblies and chopping of fuel elements into modest-sized pieces, 4) the leaching of the fuel from the cladding by nitric acid, and 5) the preparation of the acidic feed solution for the solvent-extraction separations process.

The Purex process steps are 1) solvent extraction which effects the separation from fission products and partition of uranium and plutonium, 2) purification of uranium and plutonium to separate these products from the small quantity of residual fission products, and 3) conversion of uranium to UF_6 and plutonium to PuO_2 . In addition, the waste management operations stemming from general plant operations, such as maintenance, decontamination, supporting analytical services, the general recycle and makeup plant liquids and reagents, and other such activities, are included. Figure 2.17 identifies the various steps in the reference process.

2.4.1.1 Head End Operations (Figure 2.18)⁽¹⁻³⁾

Head end operations in the fuel reprocessing plant consist of washing of the casks to remove external radioactivity and transfer of the fuel assemblies from the cask into the pool racks for storage. The cask decontamination solutions, the pool water cleanup backwash and regeneration solutions, and spent pool water filters and resins constitute a portion of the waste streams from the head end. After sufficient cooling, the fuel assemblies are chopped and the fuel fragments treated with nitric acid to produce a solution of spent

FIGURE 2.17. Reprocessing of LWR Fuel

FIGURE 2.18. Waste Generation Flow Sheet for Fuel Reprocessing
I. Head-End

fuel and a waste stream of cladding hulls and other metallic fuel components. The dissolver off-gas contains noble gas, tritium, and iodine fission products. The feed solution for solvent extraction is prepared by clarification of the dissolver solution to remove suspended solids. These solids, in the form of a sludge of insoluble radioactive materials, are part of the head end waste.

In the reference process, fuel is received from the reactors in shielded casks shipped either by rail or by truck. The receiving operations consist largely of examination of the cask and, if necessary, decontamination of its exterior. The cask is then introduced into the fuel pool, where the top is removed and individual spent fuel assemblies (see Section 2.2) are withdrawn from the cask and placed in storage racks. These are underwater procedures. Section 2.3 discusses fuel-pool wastes.

Following acceptable decay, a fuel assembly is introduced into the main portion of the reprocessing plant and disassembled. Some disassembly, e.g., removal of the shroud, may be done while the assembly is at the reactor, in the pool, or prior to the chopping. Disassembly to remove end hardware by unbolting, sawing, or shearing would remove massive metal from the fuel-containing portion, thus effecting a 10 to 15% reduction of metal waste mass.^(2,3) Alternatively, the whole assembly could be chopped, reducing the number of waste types and the number of separate operations.⁽¹⁾ In the reference case described here, all metallic components of the fuel assembly are assumed to constitute a single waste stream of hulls and hardware.

After chopping in a mechanical shear, the fuel pieces are dropped into a dissolver vessel. In the dissolver, the chopped pieces of fuel (1 to 8 cm long) are treated with concentrated nitric acid* bringing into solution most of the oxide fuel and leaving the Zircaloy, stainless steel, and Inconel essentially unattacked. In the dissolution, the noble gas fission products, some of the fission product iodine, and some of the tritium** are introduced into the vapor space above the dissolver and, hence, the dissolver off-gas system (DOG).

Following the dissolution of the fuel, the resulting solution is clarified, generally in a centrifuge, to prepare the feed for the subsequent solvent extraction operations. The clarification of the dissolver solution produces a modest amount of insoluble, highly radioactive sludge which contains fission products and may be contaminated by plutonium and actinide elements.

*Possibly containing a soluble neutron poison such as Gd, B, or Cd salts.

**Tritium distribution is shown in Figure 2.25.

2.4.1.2 Solvent Extraction Separation (Figure 2.19)^(1,4,5)

The co-decontamination and partition parts of the solvent extraction process effect transfer, initially, of uranium and plutonium into the organic phase leaving in the nitric acid solution small amounts of uranium and plutonium, and almost all of the fission product activity. This solution is the high-level waste. The partitioning portion of the flow sheet effects the separation of uranium and plutonium by selective transfer of plutonium into nitric acid. Subsequent transfer of uranium into a nitric acid phase completes the partitioning. The resultant three streams are the high-level waste in nitric acid, plutonium in nitric acid, and uranium in nitric acid. The waste streams from this portion of the reprocessing operation consist of the high-level wastes, the vessel vent off-gases, the solvent cleanup residues, and the spent solvent combustion wastes.

FIGURE 2.19. Waste Generation Flow Sheet for Fuel Reprocessing II. Solvent Extraction

The aqueous solution from the feed clarification step is contacted with an organic solution of tributyl phosphate in normal dodecane. The apparatus used is either a centrifugal contactor or a pulse column. The solvent extraction conditions are adjusted to transfer uranium and plutonium into the organic phase with a minimum of transfer of the fission products. Separation of the organic phase from the aqueous phase effects the first separation of fission products from uranium and plutonium. This co-decontamination step removes from uranium and plutonium more than 99% of the fission products in the feed solution and produces the high-level liquid (HLLW) waste stream. During these operations, the vessel off-gas system (VOG) retains sufficient radioactivity to require treatment.

Following the co-decontamination step, the organic phase containing uranium and plutonium is contacted with nitric acid containing chemicals that effect transfer of plutonium back to the aqueous solution while uranium is maintained in the organic phase. Separation of the organic and aqueous phases produces a stream of nitric acid containing plutonium and an organic phase containing uranium. The final step in the partitioning is the stripping of uranium from the organic phase into nitric acid.

It is necessary to clean the organic extractant and solvent because the high radiation from the fission products, particularly in the co-decontamination step, results in decomposition products detrimental to the extraction operations. The solvent cleanup system employs a process in which the organic phase is treated alternately with a dilute Na_2CO_3 solution and with nitric acid to remove decomposition products. The waste stream from these steps is one of the components of the intermediate-level wastes described below. Also, extensive recycle of the solvent may result in a fraction of the solvent being no longer usable in extraction operations. This material can either be absorbed in a solid or can be burned. The reference flow sheet employs the combustion of spent solvent. The residues from the treatment of the off-gas from burning of spent solvent also constitute a waste stream to be described below.

2.4.1.3 Uranium Purification and Conversion (Figure 2.20)^(1,2)

The uranium purification process involves solvent extraction, using the same general technology as applied in the co-decontamination and partitioning steps. The wastes from these operations are largely solvent cleanup wastes

and vapor treatment wastes from the vessel off-gas system. A final uranium purification step, consisting of contact of the uranium solution with silica gel, produces a waste stream of regenerant solution for the silica gel bed and, infrequently, the used silica gel. The conversion process to produce UF_6 from uranium in nitric acid consists of dehydration, denitration, and reduction steps followed by conversion of the oxide to the volatile uranium hexafluoride. The waste streams from these operations consist of off-gases which are largely not radioactive, solids from off-gas scrubbing, solids consisting of the fluorinator residue and fines in which the main portion of the radioactive contaminants can be found, and particulate filters.

The aqueous nitric acid solution of uranium obtained in the partitioning step contains not only uranium but sufficient fission products to require further decontamination before conversion to uranium hexafluoride. Uranium is further purified by repetition of the solvent extraction processes previously described for co-decontamination and partitioning. Nitric acid solutions of uranium are contacted under appropriate conditions with a solution of tributyl phosphate in normal dodecane to transfer uranium from the aqueous to the organic phase while leaving in the aqueous phase many of the fission product impurities. The aqueous phase then represents a waste stream containing fission products which will be combined with other fission product waste streams. Adjustment of conditions allows for the transfer of uranium back to a clean aqueous phase. Repetition of the transfer of uranium between aqueous nitric acid and organic tributyl phosphate solutions removes residual fission products and produces a uranium solution of sufficient purity to be used as feed for the conversion to uranium hexafluoride. A final cleanup of the uranium solution by contacting it with silica gel may be used in place of a cycle of solvent extraction to remove selected fission products.

The product of the purification of uranium is a concentrated solution of uranium in nitric acid. This solution is dehydrated to remove water and heated further to convert uranyl nitrate to uranium trioxide, which is reduced with hydrogen to form uranium dioxide. Uranium dioxide is converted to uranium tetrafluoride by gaseous hydrogen fluoride, and finally uranium tetrafluoride is converted to the volatile uranium hexafluoride using elemental fluorine. Several of these steps are carried out in large, fluidized-bed reactors in which gases are easily contacted with granular solids. The waste streams from these operations are generally of very modest radioactivity, largely because the feed uranium has been subjected to extensive purification. Nevertheless, waste streams include off-gases, off-gas cleanup wastes, the nonvolatile

FIGURE 2.20. Waste Generation Flow Sheet for Fuel Reprocessing
III. Uranium Purification and Product Formation

residues from the fluorination step, and miscellaneous solids such as filters and bulk fluidized-bed inert material.

2.4.1.4 Plutonium Purification and Conversion (Figure 2.21)^(1,2)

Plutonium purification and conversion are carried out by solvent extraction of plutonium nitrate to remove residual fission products, followed by the conversion of plutonium nitrate to the insoluble oxalate that is calcined to the dioxide. The waste streams from these parts of the process contain relatively little fission products. The waste streams come from the solvent extraction steps, e.g., liquid waste streams from the solvent cleanup operations, and some solid wastes from the precipitation of wastes in the conversion step. The vessel off-gas systems for the purification of plutonium and operations in the conversion steps require filtration and some scrubbing and generate wastes of filters and scrubber residues.

The nitric acid solution of plutonium obtained in the partitioning operations is also not pure enough to be used directly for the conversion to plutonium dioxide. Two additional cycles of solvent extraction are employed to remove the residual fission products. The solvent extraction system is similar to that described previously and consists of a contact of the solution of plutonium with tributyl phosphate in normal dodecane. The transfer of plutonium into the organic phase effects a separation between plutonium and fission products. Subsequent scrubbing of the organic solution under appropriate conditions additionally purifies the plutonium in the organic phase. A change in extraction conditions results in transfer of plutonium from the organic phase to the aqueous phase. This transfer also effects an additional purification from fission products. Repetition of this procedure for a third plutonium cycle produces a nitric acid solution of plutonium (Table 2.7) of adequate purity to be used in the preparation of plutonium dioxide.

Plutonium nitrate solution is converted to solid plutonium dioxide by formation of an insoluble plutonium oxalate, subsequent filtration of the plutonium oxalate, and calcination of the oxalate to the oxide. The conversion of plutonium nitrate to plutonium dioxide generates relatively little waste since much recycle is normally done to ensure low losses of the valuable fissile plutonium. Scrap recovery involves use of fluoride for the dissolution of plutonium dioxide, and additional purification is needed to make the plutonium solution suitable for recycle. The solvent extraction purification used for scrap recycle produces a modest amount of waste, largely aqueous. Other impurities also are present and are removed from aqueous waste solutions by precipitation with caustic. The major waste streams from the conversion of plutonium nitrate to plutonium dioxide, when carried out in conjunction with operations of the fuel reprocessing plant, are filter cakes from the caustic precipitation of impurities. Most of the aqueous solutions are recycled either

FIGURE 2.21. Waste Generation Flow Sheet for Fuel Reprocessing
IV. Plutonium Purification and Product Formation

within the conversion operations or into the solvent extraction portion of the separations plant. The vessel off-gas stream is scrubbed, producing aqueous solutions that also contain a modest amount of solids.

TABLE 2.7. Typical Plutonium Nitrate Specifications⁽¹⁾

Property or Impurity	Value
Plutonium concentration, g/liter	100-300
Nitric acid concentration, <u>M</u>	2-10
Entrained organic solvent, vol% max	0.3
Insoluble residue, ^(a) ppm Pu, max	5000
Uranium, ppm Pu, max	5000
Americium, ppm Pu, max	2800 ^(b)
Other metals, ppm Pu, max (total)	5000
Sulfur (as sulfate), ppm Pu, max	1000
Fluorine + chlorine, ppm Pu, max (total)	150
Equivalent boron content, ppm Pu, max (excluding uranium and americium)	10
Fission products	
Gamma-emitting, ^(c) $\mu\text{Ci/gPu}$	40
Zirconium-niobium-95, $\mu\text{Ci/gPu}$	5

- The insoluble residue after filtering the plutonium nitrate solution through a 100 micron filter and washing with 8 M nitric acid for one hour at 25°C.
- As determined within 120 days from time of separation.
- Gamma-emitting fission products whose parent isotopes have a half-life of 30 days or greater.

2.4.1.5 General Process Functions (Figure 2.22)^(1,2,5)

There are other operations carried out in the reprocessing plant not easily ascribable to any single portion of the process. These include general maintenance and maintenance of specific pieces of equipment, decontamination of areas and equipment, analytical services, and the recycle and preparation of reagents and solutions. The maintenance, testing, and replacement of facilities that reduce the radioactivity in effluents also generates waste streams. Maintenance involves broad, process-wide activities and generates waste streams that include filters, noncombustible trash, and some liquids. Analytical wastes are both liquid and solid, while decontamination solutions are mostly aqueous liquids that may include some organic materials. Combustible trash, spent equipment too large to be conveniently stored or disposed of in the standard 55-gal (0.2 m³) drum, and materials such as the vapors from concentrator operations are part of the wastes generated in plant-wide operations. Each of these waste streams will be described in subsequent sections.

FIGURE 2.22. Waste Generation Flow Sheet for Fuel Reprocessing
V. Process Support

2.4.2 Organization and Descriptions of Waste Streams (Figures 2.23 and 2.24) (1,2,5)

Waste streams from reprocessing operations were arbitrarily divided into five categories. The high-level waste, as defined in the Code of Federal Regulations,⁽⁷⁾ consists of the raffinate from the first solvent extraction cycle. In normal practice, the low-level aqueous waste concentrator residues and the solids from the feed clarification step are added to the high level waste stream.

A second waste stream, called the intermediate-level liquid waste, contains the off-gas treatment system (DOG, VOG) scrubber solutions. The intermediate-level waste stream also contains the pool water cleanup backwash and regeneration solutions, the cask and plant decontamination solutions, the solvent wash aqueous waste, the liquid waste generated by maintenance operations, and the liquid wastes generated by the laboratories. All of these constitute feed for the intermediate-level liquid waste treatment system.

A third waste stream is the low-level liquid waste system, which consists of the second and third uranium and plutonium cycle aqueous wastes, the condensed overhead stream from the concentrators, the condensed stream from the vessel off-gas system and the high-level waste vent system, the condensed overhead stream from conversion steps, and the aqueous stream from the plutonium purification in the conversion process. These waste streams constitute the feed to the low-level liquid waste processing operation described subsequently.

A fourth waste is the gaseous stream which is effluent from the off-gas treatment systems. The feed to the off-gas treatment systems consists of the dissolver off-gas stream, the general vessel off-gas stream, and other ventilation streams that are routed through chemical and physical retention systems in the off-gas treatment trains.

The last waste category is the solid waste. This stream includes hulls and assembly hardware, filters, failed equipment, zeolite, silica gel and resin beds, fluorinator ash and fines scrubber solids, and noncombustible trash. The reference plant processes convert all combustible TRU waste to ash, but volumes of unburned trash are given in subsequent tables. Hulls will be treated as a separate category.

2.4.2.1 High-Level Waste⁽¹⁻³⁾

The high-level waste (HLW) system receives initially the high-level aqueous stream from the co-decontamination of uranium and plutonium from the fission products contained in the dissolver solution. More than 99% of the fission products is present in this stream in relatively dilute form as generated in the co-decontamination solvent extraction step. The high-level waste from the co-decontamination contactor is evaporated to increase the concentration of fission products and to recover overhead both water and nitric

FIGURE 2.23. Waste Generated in Fuel Reprocessing Operations I. High Level, Intermediate Level, and Low Level Liquid Waste

FIGURE 2.24. Waste Generated in Fuel Reprocessing Operations
11. Gaseous and Solid Waste

acid. The concentrated high-level waste in nitric acid is routed to cooled tanks for storage. There slurries are added from the centrifuge sludge formed in the feed preparation step and the residue left in the low-level liquid waste concentrator. Liquid HLW is stored in cooled, shielded tanks because of its heat and radiation emissions.

The composition of liquid HLW as stored is shown in Table 2.8 and 2.9. The data⁽⁶⁾ are based on transfer of 99+% of the fission products to the HLW; no noble gas fission products, 0.1% of the iodine, 8% of the tritium, 0.5% of the plutonium, and 0.5% (at 160 days cooling after discharge from reactor core) of the uranium are included. The fuel burnup is assumed to be 25,000 MWd/MTHM at a specific power of 35 MW/MTHM. The data are shown for a solution with a waste volume ratio of 100 gal (378 liters)* per MTHM. Although the HLW is produced at the rate of ~5000 liters/MTHM, the rate is strongly dependent on the details of the reprocessing plant flow sheet. Storage in concentrated form is contemplated for economic reasons, i.e., greater recycle of nitric acid and water. It is likely that a part of the constituents are in the form of suspended solids.

2.4.2.2 Intermediate-Level Wastes^(1-3,5)

The intermediate-level waste (ILW) treatment system (see Figure 2.23) accepts the mercuric nitrate scrubber solutions from the dissolver off-gas and the vessel off-gas systems, incinerator scrubber solutions, the pool water cleanup backwash solutions, the aqueous solutions from cask and plant decontamination operations, the aqueous waste solutions from the washing of process solvent, HLW demister scrubber solutions, liquid wastes from laundry and laboratory operations, and process cell sump wastes. This group of wastes is neutralized and concentrated in an evaporator. The concentrated solution is transferred to a storage tank where salts gradually crystallize from the solution while the supernatant liquid is returned to the concentrator for additional boil-down. The ILW that is stored consists of the salt cake in a shielded tank. An alternative to storage of salt cake, but not included here, is to combine the ILW with the HLW prior to solidification.

The composition of the neutralized ILW solutions is described in Table 2.10.

2.4.2.3 Low-Level Liquid Waste^(1-3,5)

The low-level waste treatment system of the reprocessing plant accepts the aqueous solutions from second and third cycle uranium and plutonium purification streams, the condensate from concentrators and vessel off-gas systems, relatively clean conversion solutions, and the aqueous wastes from the washing of solvent used in the recycle of plutonium in the conversion process.

*Interim storage at 600-1100 liters/MTHM may be practical (see Section 6.0).

TABLE 2.8. Composition of Liquid High Level Waste^(a)

Basis: 378 Liter/MTHM, 25 Gwd/MTHM, 35

MW/MTHM, SGR = 1.0 for MOX Content;^(b)No Xe, Kr, 0.1% I, Br, 8% ³H,^(c) 0.5% Pu,

0.5% U, All Np, Am, Cm

Fission products ^(d)	g/Liter	Ci/Liter		Thermal W/Liter	
	[5 year] ^(e)	1 year ^(f)	10 year ^(f)	1 year ^(f)	10 year ^(f)
³ H ^{(c)(g)}	8 E-6 ^(h)	0.10	0.06	----	----
Rb	0.8	----	----	----	----
Sr	1.5	140	110	0.16	0.13
Y	0.86	150	110	0.80	0.62
Zr	7.0	17	----	0.08	----
Nb	----	38	----	0.18	----
Mo	6.9	----	----	----	----
Tc	1.7	0.03	0.03	----	----
Ru	4.5	540	1.1	0.03	----
Rh	1.1	540	1.1	5.30	0.01
Pd	3.1	----	----	----	----
Ag	0.16	52	----	0.38	----
Cd	0.18	0.07	0.04	----	----
Sn	0.11	0.83	----	----	----
Sb	0.03	16.8	1.7	0.05	----
Te	1	7.2	0.42	----	----
Cs	5.4	470	190	2.80	0.31
Ba	3.3	200	160	0.80	0.65
La	2.5	----	----	----	----
Ce	4.8	820	0.3	0.54	----
Pr	2.3	830	0.3	6.29	----
Nd	8.0	----	----	----	----
Pm	0.08	210	19.5	0.08	----
Sm	1.8	2.9	2.7	----	----
Eu	0.3	21.7	9.7	0.16	0.08
Gd	0.2	----	----	----	----
Total F.P. ⁽ⁱ⁾	57.75	4074	614.6	17.7	1.81
<u>Actinides</u>					
U	12.65	----	----	----	----
Np	0.674	0.36	0.36	----	----
Pu	0.4	3.96	3.03	0.03	0.03
Am	2.3	2.07	2.07	0.06	0.06
Cm	0.65	99.7	45.8	3.56	1.60
Total Act. ⁽ⁱ⁾	16.79	106.2	51.3	3.65	1.69
Total	74.5	4180	664	21.35	3.50
<u>Chemicals</u>					
HNO ₃ ^(k)	126	----	----	----	----
Gd ^(k)	23.8	----	----	----	----
PO ₄ --- ^(m)	4	----	----	----	----
Fe	3	----	----	----	----
Cr	0.5	----	----	----	----
Ni	0.2	----	----	----	----
Na ^(m)	0.1	----	----	----	----

Footnotes follow after next page.

TABLE 2.9. Composition of High Level Waste^(a)

Basis: 25 GWd/MTHM, 35 MW/MTHM,
 SGR = 1.0 for MO Content^(b) No Xe,
 Kr, 0.1% I, Br, 8% ³H,^(c) 0.5% Pu,
 U, All Np, Am, Cm

Fission products ^(d)	g/MTHM	Ci/MTHM		Thermal W/MTHM	
	[5 year] ^(e)	1 year ^(f)	10 year ^(f)	1 year ^(f)	10 year ^(f)
³ H ^{(c)(g)}	3 E-3 ^(h)	4.0 E1	2.4 E1	----	----
Rb	3.1 E2	----	----	----	----
Sr	5.8 E2	5.4 E4	4.3 E4	6.2 E1	5 E1
Y	3.2 E2	5.5 E4	4.3 E4	3.0 E2	2.3 E2
Zr	2.6 E3	6.4 E3	2.1	3.2 E1	----
Nb	----	1.4 E4	----	6.8 E1	----
Mo	2.6 E3	----	----	----	----
Tc	6.5 E3	1.1 E1	1.1 E1	----	----
Ru	1.7 E3	2.05 E5	4.2 E2	1.3 E1	----
Rh	4.3 E2	2.05 E5	4.2 E2	2 E3	4.1
Pd	1.2 E3	----	----	----	----
Ag	6.2 E1	2 E4	----	1.4 E2	----
Cd	6.8 E1	2.5 E1	1.6 E1	----	----
Sn	4.1 E1	3.2 E2	----	----	----
Sb	1.1 E1	6.4 E3	6.5 E2	2 E2	----
Te	3.8 E2	2.7 E3	1.6 E2	----	----
Cs	2.0 E3	1.8 E5	7.1 E4	1.1 E3	1.2 E2
Ba	1.2 E3	7.7 E4	6.2 E4	3.0 E2	2.4 E2
La	9.4 E2	----	----	----	----
Ce	1.8 E3	3.1 E5	1.0 E2	2.1 E2	----
Pr	8.7 E2	3.1 E5	1.0 E2	2.4 E3	----
Nd	3.0 E3	----	----	----	----
Pm	3 E1	8.0 E4	7.4 E3	3 E1	2
Sm	6.9 E2	1.1 E3	1.0 E3	----	----
Eu	1.1 E2	8.2 E3	3.7 E3	6.1 E1	2.9 E1
Gd	6.6 E1	----	----	----	----
Total F.P. ⁽¹⁾	2.18 E4	1.54 E6	2.32 E5	6.69 E3	6.85 E2
<u>Actinides</u>					
U	4.8 E3	4 E-2	3 E-2	2 E-4	4 E-4
Np	2.5 E2	1.4 E2	1.4 E2	1.9 E-1	1.9 E-1
Pu	1.5 E2	1.5 E3	1.1 E3	9.7	1.2 E1
Am	8.7 E2	7.8 E2	7.8 E2	2.3 E1	2.3 E1
Cm	2.4 E2	3.8 E4	1.7 E4	1.3 E3	6.1 E2
Total Act. ⁽¹⁾	6.25 E3	4.02 E4	1.94 E4	1.38 E3	6.4 E2
Total	2.82 E4	1.58 E6	2.51 E5	8.07 E3	1.32 E3
S.F. n/sec MTHM ^(j)		3.64 E9	2.49 E9	---	----
<u>Chemicals</u>					
HNO ₃ ^(k)	4.76 E4	----	----	----	----
Gd ^(k)	9 E3	----	----	----	----
PO ₄ ^(l)	1.5 E3	----	----	----	----
Fe ^(l)	1.13 E3	----	----	----	----
Cr	1.89 E2	----	----	----	----
Ni ^(j)	7.6 E1	----	----	----	----
Na	3.8 E1	----	----	----	----

Footnotes follow on next page.

Footnotes to Tables 2.8 and 2.9

- a. Data from Reference 6 unless otherwise indicated.
- b. Data from Reference 9. See Description of Fuel Assembly in Section 2.3; SGR = Self-Generating Ratio. Percentages apply to nuclide composition of fuel at 160 days after removal from core.
- c. Reference 5.
- d. Selected fission products with significant mass, activity, or thermal power.
- e. Based on mass of element after 5 year, 160 days cooling of waste after discharge from reactor core.
- f. After cooling for indicated period following processing at 160 days, 1 year cooling corresponds to 1.4 year elapsed time from discharge from core.
- g. Modified, for greater yield from ^{239}Pu fission, to 547 Ci/MTHM at discharge.
- h. 11 E-6 read as 11×10^{-6} , 5.8 E2 read as 5.8×10^2 , etc.
- i. Total for all elements in the category, listed or not.
- j. Spontaneous fission neutrons only; >99% due to Cm and Cf.
- k. From Reference 10, waste category PW-7; HNO_3 is 2 M free acid. Gd added as a neutron poison.
- la. Based on Reference 1 flow sheet.

TABLE 2.10. Properties of Intermediate Level Waste Solution After Neutralization

	<u>g/liter^(a)</u>	<u>Ci/year</u>	<u>kg/year^(a)</u>
<u>Chemical Composition</u>			
NO ₃ ⁻	400	--	66,000
Na	150	--	24,750
Hg	15	--	2,475
Ca, Mn, SO ₄ ⁻ , Cl	1 each	--	165 each
K, Fe, Ag, F	0.5 each	--	82 each
Misc. Chem. ^(b)	minor	--	----
<u>Radioactivity</u>			
U	20	--	3,300
Pu	0.05	--	8.25
I ^(b)	1.8	2300	297
Fission Products ^(d)	trace	3300	----
<u>Physical Properties</u>			
Volume, liter/year ^(e,f)		165,000	
Solids ^(a)			97,000
Density (estimate)	1400		
Heat Generation (W/l)		0.01-0.1	

a. Based on ~42% dissolved solids.

b. Oxalate, citrate, phosphate, acetate, ammonium, chromate, surfactants, detergents, etc.

c. Based on ~100% of iodine in this stream, 79% ¹²⁹I, 21% ¹²⁷I by weight, and 98% ¹³¹I, 2% ¹²⁹I by activity, at 320 days following discharge from reactor core.

d. Based on 2.2 Ci/MTHM. These values may be low by a factor of 10 or more, compared to values based on other estimates. Activity contributed primarily by fission product Ru, Rh, Zr, and Nb.

e. Based on 110 liters per MTHM, 1500 MTHM per year.

f. It is estimated that spent, combustible solvents₃ will be generated at a FRP at the rate of 0.01 m³/MTU or 15 m³/1500 MTU-year,⁽⁸⁾ but the reference process includes combustion of the solvent.

This waste is concentrated in a general purpose concentrator or in a low level waste concentrator and the overheads from this operation are usually vented in vapor form directly to the stack. The very low concentration of radionuclides in this waste is shown in Table 2.11. The major contaminant is tritium, present as a dilute solution of HTO in H₂O.

TABLE 2.11. Estimated Composition of Low Level Liquid Waste^(a)

Radionuclides	$\mu\text{Ci}/\text{cm}^3$	Chemicals	ppm (wt)
^3H	10	Na	1 E-7
^{90}Sr	6 E-7	Fe	3 E-6
^{106}Ru	3 E-6	Cr	1 E-7
^{131}I	5 E-6	PO_4^{3-}	1 E-6
^{129}I	6 E-7	U	1 E-6
^{137}Cs	8 E-7		
^{144}Ce	6 E-6		
Pu (α)	3 E-8		
^{241}Pu	1 E-6		
^{241}Am	2 E-9		

a. Bases: LLLW after evaporation; nuclide and chemical content $\sim 1.0 \text{ E-10}$ of HLW, Volume is 3.8 E4 liter per MTHM; nitric acid is 0.01 M (600 ppm); tritium content $\sim 72\%$ of charge; about 180 days cooled.

The distribution of tritium in the various reprocessing waste streams is based on assumptions illustrated in Figure 2.25. Data on the distribution of tritium in spent LWR fuel and processing waste streams is relatively incomplete.⁽⁵⁾ The assumed distribution shows the majority of the tritium in the low level liquid waste stream which is in general agreement with experience. The data in Figure 2.25 are for no decay (cooling) following discharge of the reference fuel.

The residue remaining in the concentrator is transferred to the high level waste tanks for storage. No data are presented for this stream since further treatment of the contained waste is included in the description of high level waste.

2.4.2.4 Gaseous Waste^(1-3,5)

The vapors from the dissolver off-gas system and the vessel off-gas system are passed through scrubbers of mercuric nitrate to remove iodine* from the off-gas stream. They are subsequently filtered for additional iodine removal in silver zeolite beds before final filtration through HEPA filters.

*Long-lived ^{129}I is expected at the rate of $2 \times 10^2 \text{ g}$ ($3 \times 10^{-2} \text{ Ci}$)/MTHM or $3 \times 10^5 \text{ g}$ (50 Ci)/1500 MTHM-year.

FIGURE 2.25. Distribution of Tritium in Fuel Reprocessing Wastes
(100% ³H Assumes No Cooling of Reference Fuel)

The treated gases are vented to the stack. The off-gas treatment system does not generate gaseous wastes that need to be retained. All the fission product krypton and xenon is discharged by this path. Table 2.12 shows the expected quantities of noble gas fission products. The filters and solid iodine absorbing materials are treated as a solid waste, and the iodine-containing scrubber solutions are treated in the intermediate-level waste treatment system.*

The presence of ^{14}C in LWR fuel is the result of the $^{14}\text{N}(n,p)^{14}\text{C}$ reaction with the nitrogen impurity in the fuel and, to a lesser extent, the reaction $^{17}\text{O}(n,\alpha)^{14}\text{C}$ with the oxygen of the fuel ceramic. The carbon impurity in fuel represents a negligible source of ^{14}C since the $^{13}\text{C}(n,\gamma)^{14}\text{C}$ reaction has a low cross section. Estimates of ^{14}C production in LWR fuel vary considerably and no definitive analyses have been reported.⁽¹⁵⁾ Further, the movement of ^{14}C through the fuel cycle has not been defined. For the present discussion, it is assumed that 25 ppm nitrogen impurity results in a ^{14}C production of 20 Ci/1000 MWe-year, and that all of the ^{14}C will be found in the gaseous waste as CO_2 .** These data are equivalent to 0.4 Ci ^{14}C per MTHM and 660 Ci ^{14}C per year from a reprocessing plant with a 1500 MTHM/year throughput.

2.4.2.5 Solid Wastes (1-3,17,18)

The solid waste generated in fuel reprocessing comprises several streams. The hulls and fuel assembly hardware from head end operations are described in a subsequent section. A class of solid wastes consisting of wet solids includes filters used to filter the pool water, ion exchange resins used to purify the pool water, silica gel from the final stage of uranium purification, and the filter cake from the liquid waste treatment used in converting plutonium nitrate to plutonium oxide. The bulk of the CaF_2 sludge from conversion process scrubbers is nonradioactive. Another source of solid wastes is the failed process equipment of varying size, normally at least partially decontaminated before removal from the process enclosures. Storage of large, contaminated equipment is in special containers or in a failed equipment vault. Additional sources of solid wastes include the noncombustible trash, the ash from the uranium fluorination, ventilation filters such as the HEPA filters, and the ash from combustors.[†]

*If the ILW is to be combined with the HLW and calcined, mercury-containing scrubber solutions need to be handled separately, e.g., as a waste formed by sorbing the solution onto a solid.

**It is recognized that neither the anticipated chemistry of ^{14}C nor the behavior of CO_2 would result in a single result as outlined.

[†]The reprocessing plant is designed to burn solvent no longer needed as well as some of the combustible radioactive wastes that are generated.

TABLE 2.12. Noble Gas Fission Products Available
in Fuel Reprocessing^(a)

	Kr ^(d)	Xe
g/MTHM ^(b)	250	4061
kg/year ^(a)	375	6091
m ³ (STP)/MTHM ^(c)	0.066	0.678
m ³ (STP)/year	99	1017
Ci/MTHM		
90 days cooled	6860	127
120 days cooled	6827	20
1 year cooled	6543	10 ⁻⁵
10 years cooled	3652	----
MCi/1500 MTHM		
1 year cooled	9.81	----
10 years cooled	5.48	----
W/MTHM		
90 days cooled	10.3	0.11
120 days cooled	10.2	0.02
1 year cooled	9.8	----
10 years cooled	5.5	----
kW/1500 MTHM		
1 year cooled	14.7	----
10 years cooled	8.2	----

- a. Based on 25,000 MWd/MTHM, 35 MW/MTHM; MOX mixture based on self-generating ratio = 1.0 using 30 wt % MOX at 4.98 wt % Pu in natural uranium and 70% 3.3% ²³⁵U in uranium; plant throughput 1500 MTHM/year. 1 year cooled off-gas will also contain 26 Ci tritium per MTHM (5% of contained tritium in fuel) as HT, and 0.4 Ci ¹⁴C/MTHM as CO₂.
- b. Estimated at 5 years cooled after processing 160 days following discharge from core.
- c. Atomic weight of Kr = 84.8, Xe = 134.2.
- d. Krypton will contain ~6% ⁸⁵Kr. The total krypton will be ~120 ppm of the total dissolver off-gas.

Other solid wastes are the zeolite beds used to absorb iodine from the off-gas system, and the ash from incinerated combustible wastes. Most of these solid wastes are stored in relatively leak-free containers, particularly if it seems they might contain significant quantities of transuranium elements.

Table 2.13 lists the solid waste streams expected from reprocessing operations, using the best data currently available. The magnitudes of volumes, mass, and contamination levels are closely related to the details of selected process steps, plant operation philosophy, and other factors. The data in Table 2.13 are restated in summary form in Table 2.14 for the reference 1500 MTHM/year plant. Selection of contamination designations for waste was based on the assumption that intermediate levels of beta-gamma contamination are usually associated with TRU contamination. Further, the burning of spent solvent at an estimated rate of ~ 8 kg/hr,⁽¹⁴⁾ produces a waste stream that is mixed with ILW salts and stored, while some combustible LL, non-TRU waste is reduced to a dry, solid ash.

2.4.2.6 Hulls and Fuel Assembly Hardware^(1-3,11-13,19)

Chopped hulls are rinsed and removed from the dissolver after dissolution of the oxidic fuel by nitric acid. Massive end pieces and internal hardware are also part of this stream. Zircaloy fines generated by sawing and chopping are oxidized prior to mixing with the hulls stream. Tables 2.15 and 2.16 summarize the properties of the metallic waste in the hull stream. Data in these tables can be converted to other units (e.g., per GWe-year, per fuel assembly) using the conversion factors in the footnotes. The metallic waste is assumed to contain 0.05% of the initial fuel,* almost entirely in the form of unleached oxidic fuel in the cladding hulls. Further, the hulls are assumed to contain 15% of the tritium formed in the fuel. The volume of the metal waste is estimated based on an average bulk density of uncompacted hulls and hardware of 1 g/cm^3 . Oxidized Zircaloy fines represent an insignificant mass or volume contribution to this stream. The metallic waste is packaged in containers of high integrity after being surveyed for excessive amounts of undissolved reactor fuel. Data in Tables 2.15 and 2.16 do not include fuel assembly channels. It is assumed that these are reused at the reactor and, when no longer serviceable, are discarded as a solid waste generated at the reactor. Amounts of fuel channel waste are estimated to be equivalent to 100 kg Zircaloy per MTHM or about 3.3 MT Zircaloy per 1000 MWe-year.⁽¹⁶⁾ This waste may have a greater density than hulls and may represent a volume of 1 m^3 /1000 MWe-year.

2.4.3 Alternative Fuel Cycles

As described in Section 2.1, alternative fuel cycles have an impact on the nature of the waste generated in reprocessing operations. The "throwaway" fuel cycle produces a waste consisting of the entire fuel assembly. The fission product and actinide content of the assembly, listed in Table 2.17, differs from high-level waste generated in processing of the MOX reference fuel

*Early experience suggests that fuel contamination could be as high as 0.1%.⁽¹²⁾

TABLE 2.13. Solid Wastes From Fuel Reprocessing Operations

Category	Type	Contents	Class (a)	Quantities			
				per MTHM (c)		Annual (b)	
				Mass	Volume (m ³)	Mass	Volume (m ³)
WET SOLIDS	K ₂ UO ₄ mud	K ₂ UO ₄ (1 kg U/MTHM), KOH, H ₂ O	LL, non-TRU	80 kg	0.05	120 MT	75
	Ion exchange resins, silica gel	Organic resins, silica gel, 100 Ci/m ³ mixed f.p., Pu, up to 15%.	IL, non-TRU	25 kg	0.05	37.5 MT	75
	Filter aid, sludge	Free H ₂ O, sand, diatom., earth, flocs, etc.	IL, TRU	5 kg	0.01	7.5 MT	15
DRY SOLIDS	Fluorinator fines	CaF ₂ , UF ₄ (6 g U/MTHM), PuF ₄ (10 mg Pu/MTHM), fission product fluorides (0.6 Ci/MTHM).	LL, TRU	50 kg	0.1	75 MT	150
	Fluorinator ash						
	Combustor ash	Mixed fission products.	LL, TRU	----	0.01	----	15
	Ag zeolite	Zeolite, fission product iodine.	IL, TRU	5 kg	0.005	7.5 MT	7.5
COMBUSTIBLE TRASH	Cellulosic, clothes, rubber, sorbents, etc.	Misc. organic; mixed f.p.; actinides: (IL-TRU 80 mg Pu/m ³).	IL, TRU ^(d)	----	0.7	----	1050
			LL, TRU ^(d)	----	0.3	----	450
			LL, non-TRU ^(d)	----	0.7	----	1050

TABLE 2.13. (contd)

Category	Type	Contents	Class (a)	Quantities			
				per MTHM ^(c)		Annual ^(b)	
				Mass	Volume (m ³)	Mass	Volume (m ³)
NON-COMBUSTIBLE TRASH	Small equipment, tools, glass	Metals, ceramics.	IL, TRU	----	0.7	----	1050
			LL, TRU	----	0.2	----	300
			LL, non-TRU	----	0.1	----	150
EQUIPMENT	Process vessels, apparatus	Metals, inorganic, materials.	IL, TRU	----	0.1	----	150
			LL, non-TRU	----	0.1	----	150
FILTERS	HEPA, roughing filters	Metal, glass, organics.	IL, TRU	----	0.06	----	90
			LL, TRU	----	0.1	----	150
			LL, on-TRU	----	0.01	----	15

a. Class designation based on "200 mr/hr" cut-off for LL/IL and 10 nCi Pu/g for non-TRU/TRU. Class IL usually contains mixed fission products and, except for storage pool resins, will also be a TRU waste.

b. Based on 1500 MTHM/year plant capacity.

c. MTHM = metric ton of heavy metal (U + Pu).

d. Uncompacted, noncombusted form.

TABLE 2.14. Summary of Solid Waste from Fuel Reprocessing

	Annual Rate for 1500 MTHM/year, in m ³	
	TRU	non-TRU
Combustible (trash) ^(a)	1500	1050
Noncombustible solids	1660	315
Slurries, sludges, resins	15	150
Filters	240	15
	<hr/> ~3400	<hr/> ~1500

a. Uncompacted, noncombusted form.

TABLE 2.15. Properties of Metals and Selected Elements Contained in Fuel Assembly Waste^(a)
per 1000 kg of hulls and hardware

	Mass (g)	Activity, Ci ^(b)			Heat Generation, W ^(b)		
		120 day	1 year	10 year	120 day	1 year	10 year
Metal	1 E6	2.7 E5	1.1 E5	1.2 E4	1.5 E3	5.8 E2	8.6 E1
Zircaloy ^(g)	8.9 E5 ^(f)	----	----	----	----	----	----
Stainless Steel	8.9 E4	----	----	----	----	----	----
Inconel	2.5 E4	----	----	----	----	----	----
Mn ^(d)	1.8 E3	1.7 E4	9.8 E2	5.0 E-1	1.4 E2	7.9 E0	4.1 E-3
Fe ^(d)	6.8 E4	2.2 E4	1.8 E4	1.6 E3	2.9 E1	2.3 E1	2.1 E0
Co ^(d)	2.0 E2	2.8 E4	1.6 E4	4.4 E3	4.3 E2	2.5 E2	6.8 E1
Ni ^(d)	2.1 E4	7.8 E2	7.8 E2	7.3 E2	1.6 E-1	1.6 E-1	1.5 E-1
Sb ^(d)	8.4 E2	5.7 E4	3.5 E4	3.3 E3	2.5 E2	1.5 E2	1.4 E1
Te ^(d)	4.8 E1	1.6 E4	1.4 E4	1.4 E3	1.4 E1	1.3 E1	1.3 E0
Fission products ^(c)	3.4 E1	8.4 E3	3.2 E2	3.7 E2	3.5 E1	1.4 E1	1.1 E0
TRU Elements ^(c)	2.7 E1	5.4 E2	4.7 E2	2.9 E2	6.0 E0	3.0 E0	1.5 E0
n/sec		6.4 E6	5.9 E6	3.9 E6	----	----	----
Totals/MTHM	3.3 E5	9.1 E4	3.6 E4	4.2 E3	5.0 E2	2.0 E2	2.9 E1
Totals/MT hulls-hardware	1.0 E6	2.8 E5	1.1 E5	1.3 E4	1.5 E3	6.0 E2	8.9 E1

- a. Based on data in Table 2.2 for the reference LWR fuel assembly, 25,000 MWd/MTHM burnup and 35 MW/MTHM specific power.⁽⁶⁾
- b. For cooling times indicated.
- c. Based on residual fuel in cladding of 0.05%.
- d. Selected on the basis of significant activity fraction after more than
- e. 1000 kg of metallic waste $\approx$ 9.898 fuel assemblies $\approx$ 3.08 MTHM. A 1500 MTHM/year plant generates 487 MT of this metallic waste, or 14.8 MT of waste/1000 MWe-year. At an average density of 1 g/cm³, the metallic waste from one day operation of the 1500 MTHM/year plant has a volume of $\approx$ 1.6 m³.
- f. Includes from 90 to 900 g zirconium fines (0.01 to 0.1% of Zircaloy).
- g. Includes 15% of the tritium, corresponding to $\approx$ 77 Ci/MTHM (239 Ci³H/1000 kg hulls) at 1-year cooling.

TABLE 2.16. Radioactivity Content in Reference Fuel-Assembly Metal Waste as a Function of Long-Term Cooling^(a)

Fission products ^(b)		Cooling Periods				
		10 year	100 year	1000 year	3000 year	10,000 year
Ci/MT (metal waste)		3.7 E2	2.0 E1	2.7 E-2	2.6 E-2	2.6 E-2
Ci/MTHM (fuel)		1.2 E2	1.3 E1	8.9 E-3	8.5 E-3	8.3 E-3
Ci/GWe-year		5.5 E3	6.0 E2	4.0 E-1	3.9 E-1	3.8 E-1
Predominant fission products						
Ci/MT (metal waste) ^(c)						
	Se	----	----	4.5 E-4	4.4 E-4	4.1 E-4
	Zr	----	----	3.3 E-3	3.3 E-3	3.3 E-3
	Nb	----	----	3.1 E-3	3.1 E-3	3.1 E-3
	Tc	----	----	1.7 E-2	1.7 E-2	1.7 E-2
	Pd	----	----	1.6 E-4	1.6 E-4	1.6 E-4
	Sn	----	----	7.5 E-4	7.4 E-4	7.1 E-4
	Sb	----	----	8.6 E-4	8.5 E-4	8.1 E-4
	Cs	----	----	5.3 E-4	5.3 E-4	5.3 E-4
Activation products						
Ci/MT (metal waste)		1.2 E4	3.8 E2	7.9 E0	6.9 E0	5.6 E0
Ci/MTHM (fuel)		3.9 E3	1.2 E2	2.6 E0	2.2 E0	1.8 E0
Ci/GWe-year		1.8 E5	5.7 E3	1.2 E2	1.0 E2	8.3 E1
Predominant activation products						
Ci/MT (metal waste) ^(c)						
	Ni	----	----	5.3 E0	4.8 E0	4.5 E0
	Zr	----	----	2.3 E-1	2.3 E-1	2.3 E-1
	Nb	----	----	9.7 E-2	9.5 E-2	9.1 E-2
	Mo	----	----	1.4 E-2	1.2 E-2	7.0 E-3
	Tc	----	----	1.9 E-2	1.9 E-2	1.9 E-2

TABLE 2.16. (contd)

Actinides ^(b)	Cooling Periods				
	10 year	100 year	1000 year	3000 year	10,000 year
Ci/MT (metal waste)	2.9 E2	2.4 E1	5.4 E0	2.3 E0	1.3 E0
Ci/MTHM (fuel)	9.5 E1	7.9 E0	1.8 E0	7.4 E-1	4.3 E-1
Ci/GWe-year	4.3 E3	3.6 E2	8.0 E1	3.4 E1	2.0 E1
Predominant actinides					
Ci/MT (metal waste) ^(c)					
Pa	----	----	3.0 E-3	3.6 E-3	3.6 E-3
U	----	----	5.9 E-3	5.9 E-3	6.1 E-3
Np	----	----	1.9 E-1	1.7 E-1	8.9 E-2
Pu	----	----	2.1 E0	1.8 E0	1.1 E0
Am	----	----	3.2 E0	2.9 E-1	8.9 E-2
Gross totals (fission products, activation products, and actinides)					
Ci/MT (metal waste)	1.3 E4	4.4 E2	1.3 E1	9.2 E0	6.9 E0
Ci/MTHM (fuel)	4.1 E3	1.4 E2	4.4 E0	3.0 E0	2.2 E0
Ci/GWe-year	1.9 E5	6.7 E3	2.0 E2	1.3 E2	1.0 E2

a. Based on reference fuel assembly (Tables 2.2, 2.3, 2.4, and 2.5) and reactor performance of 25,000 MWd/MTHM burnup and 35 MW/MTHM specific power.

b. Based on 0.05% residual fuel with cladding.

c. Values for Ci/MT (metal waste) convertible by applying the following factors:

 X 0.324 = amount/MTHM (fuel)

 X 14.8 = amount/GWe-year

 X 487 = amount/1500 MTHM (fuel).

TABLE 2.17. Fission Product and Actinide Content of LWR Fuel
for a "Throwaway" Fuel Cycle^(a)

	g/MTHM ^(c)	Ci/MTHM ^(b)					
		90 day	160 day	1 year	10 year	100 year	1000 year 3000 year
Fission products							
³ H ^(d)	5.1 E-2	5.2 E2	5.1 E2	4.9 E2	3.0 E2	1.8 E0	----
Ge	3.3 E-1	----	----	----	----	----	----
As	9.5 E-2	----	----	----	----	----	----
Se	4.4 E1	3.0 E-1	3.0 E-1	3.0 E-1	3.0 E-1	3.0 E-1	3.0 E-1
Br	1.7 E1	----	----	----	----	----	----
Kr	2.9 E2	7.7 E3	7.6 E3	7.4 E3	4.1 E3	1.2 E1	1 E-6
Rb	2.8 E2	1.9 E1	1.4 E1	7 E-4	2 E-5	2 E-5	2 E-5
Sr	7.2 E2	3.5 E5	1.7 E5	7 E4	5 E4	5 E3	1 E-6
Y	3.8 E2	4.9 E5	2.5 E5	7.9 E4	5 E4	5 E3	1 E-6
Zr	2.8 E3	6.4 E5	3.1 E5	3.5 E4	2.3 E0	2.3 E0	2.3 E0
Nb	1.9 E0	1.1 E6	5.8 E5	7.5 E4	1.0 E0	2.2 E0	2.2 E0
Mo	2.6 E3	----	----	----	----	----	----
Tc	6.5 E2	1.1 E1	1.1 E1	1.1 E1	1.1 E1	1.1 E1	1.1 E1
Ru	1.6 E3	6.7 E5	4 E5	2 E5	4.4 E2	----	----
Rh	3.7 E2	6.7 E5	4 E5	2 E5	4.4 E2	----	----
Pd	8.4 E2	7 E-2	7 E-2	7 E-2	7 E-2	7 E-2	7 E-2
Ag	4.7 E1	4.4 E4	3.6 E4	2 E4	2.4 E0	1 E-5	----
Cd	4.4 E1	2.8 E2	1.1 E2	2.3 E1	1.3 E1	0.2 E0	----
In	1.0 E0	1.7 E0	0.6 E0	4 E-2	----	----	----
Sn	3.4 E1	2.8 E3	1.9 E3	6.4 E2	0.5 E0	0.4 E0	0.4 E0
Sb	1.1 E1	7 E3	6.6 E3	5.7 E3	8.8 E2	0.5 E0	0.4 E0
Te	3.4 E2	3.8 E4	1.6 E4	4.2 E3	1.4 E2	----	----
I	1.7 E2	4.2 E2	1.0 E0	2.4 E-2	2.4 E-2	2.4 E-2	2.4 E-2
Xe	4.1 E3	1.2 E2	1.9 E0	1 E-5	----	----	----
Cs	2.1 E3	2.3 E5	2.2 E5	2 E5	7.2 E4	8.3 E3	0.3 E0
Ba	1.1 E3	9.1 E4	7.8 E4	7.7 E4	6.2 E4	7.9 E3	8 E-6
La	9.6 E2	1.5 E4	3.4 E2	5 E-3	----	----	----
Ce	2.0 E3	1.2 E6	8.5 E5	4.8 E5	1.6 E2	2 E-5	2 E-5
Pr	8.9 E2	9.7 E5	8 E5	4.9 E5	1.6 E2	2 E-5	2 E-5
Nd	3.0 E3	2.3 E3	2.8 E1	7 E-5	----	----	----
Pm	9.4 E1	1.2 E5	1 E5	8.7 E4	8.1 E3	----	----
Sm	6.1 E2	9.8 E2	9.8 E2	9.8 E2	9.1 E2	4.7 E2	0.6 E0
Eu	1.1 E2	1 E4	8.3 E3	7.8 E3	3.8 E3	2.2 E0	----

TABLE 2.17. (contd)

		Ci/MTHM ^(b)						
	<u>g/MTHM^(c)</u>	<u>90 day</u>	<u>160 day</u>	<u>1 year</u>	<u>10 year</u>	<u>100 year</u>	<u>1000 year</u>	<u>3000 year</u>
<u>Fission products (contd)</u>								
Gd	5.1 E1	1.1 E1	9.1 E0	5.0 E0	----	----	----	----
Tb	1.2 E0	3.6 E2	1.8 E2	2.5 E1	----	----	----	----
<u>Actinides</u>								
U	9.7 E5	8.3 E1	3.3 E0	3.2 E0	2.7 E0	1.8 E0	2.0 E0	2.1 E0
Np	3.3 E2	6.5 E0	6.5 E0	6.5 E0	6.5 E0	6.5 E0	6.3 E0	5.5 E0
Pu	7.8 E3	7.4 E4	7.3 E4	7.1 E4	4.7 E4	2.0 E3	6.6 E2	5.8 E2
Am	7.9 E1	8.7 E1	1.1 E2	1.7 E2	9.7 E2	2.2 E3	5.3 E2	2.6 E1
Cm	8.2 E0	9.9 E3	7.5 E3	3.5 E3	4 E2	1.5 E1	1.0 E-1	6 E-2
Bk	5 E-8	1.5 E-4	1 E-4	8 E-5	----	----	----	----
Cf	9 E-8	2.9 E-6	2.8 E-6	2.7 E-6	1.5 E-6	4 E-7	----	----
Total fission products								
	2.6 E4	6.6 E6	4.2 E6	2.1 E6	2.5 E5	2.7 E4	1.8 E1	1.7 E1
Total actinides								
	9.7 E5	8.4 E4	8.1 E4	7.5 E4	4.8 E4	4.2 E3	1.2 E3	6.1 E2
Total thermal watts/MTHM								
		2.7 E4	1.9 E4	9.2 E3	8.6 E2	1.9 E2	3.8 E1	1.9 E1
Total neutrons/sec-MTHM ^(e)								
		1.7 E8	1.5 E8	1.1 E8	6.4 E7	1.1 E7	4.2 E6	2.7 E6

a. Basis: 25,000 MWd/MTHM, 35 MW/MTHM, initial ²³⁵U enrichment 3.3%. Data from Reference 6.

b. Curies of activity from the indicated elements, per metric ton of heavy metal in discharged fuel cooled for the times stated at column headings.

c. Based on element present after 1 year cooling.

d. Based on 523 Ci ³H/MTHM at discharge in fuel of 25,000 MWd/MTHM burnup and $T_{1/2} = 12.262$ year.

e. Includes (α, n) and spontaneous fission neutrons.

largely because of the absence of initial plutonium in the "throwaway" fuel and of processes that remove the bulk of the uranium and plutonium from the spent fuel. Hence, the waste from the "throwaway" fuel cycle contains 970 kg of uranium per MTHM (with 1.22% ^{235}U remaining) and 7.8 kg plutonium. Table 2.18 compares the waste composition from LWR reference fuel processing and from the "throwaway" fuel cycle.

Another fuel cycle alternative requires the reprocessing of the UO_2 fuel to segregate uranium, plutonium, and fission products but includes storage of plutonium rather than incorporation of plutonium into MOX fuel. This alternative generates all the types of reprocessing waste found in the reference fuel cycle but the composition of the wastes differs slightly from that found in the reference case. Table 2.19 contains summary data on the radiation from high-level waste obtained by reprocessing UO_2 fuel. The differences between the LWR reference high-level waste (Table 2.9) and the UO_2 reprocessing high-level waste are largely due to the actinide content of the former waste.

Data shown in Table 2.19 illustrate the activity, heat generation, and neutron emission from high-level waste from processing of UO_2 fuel for various cooling times. These data are similar to those in Table 2.17 except for the removal of 99.5% U and Pu. Comparable data for the LWR reference fuel cycle are shown in Table 2.9. Data in Table 2.20 illustrate more specifically the reduction in actinide content of the waste from processing of UO_2 fuel.

Minor differences in the distribution of actinide isotopes will also be evident in all TRU wastes from reprocessing operations. The concentration of TRU contamination of non-high-level waste is not great enough to require new source term descriptions for treatment technologies. For a constant mass contamination level, TRU waste generated by processing only UO_2 fuel should show a reduction in curie level proportional to that shown for the fuel or high level waste. In all cases, the fission product spectrum is sufficiently similar so that no consequential differences in the wastes result from the processing of UO_2 fuel.

Conversion of plutonium nitrate to plutonium dioxide is common to the LWR reference fuel cycle and to the "uranium-only" fuel cycle, in which PuO_2 is stored. Storage of PuO_2 for prolonged periods may require close attention to the extent to which volatile and decomposable materials are removed from the dioxide on calcination. No significant differences are anticipated in technology for the conversion process. Storage of PuO_2 is expected to be in small batches, e.g., 5 kg PuO_2 ,* properly packaged and spaced for criticality and radiologic safety. Since the 5 kg batches will generate ~40 W of thermal energy, large arrays may require forced cooling. The isotopic composition and

* 4.4×10^3 g of Pu.

TABLE 2.18. Properties of High-Level Waste from Processing of LWR Reference Fuel Compared with Properties of "Throw-away" Fuel^(a)

Property	"Throwaway" Fuel ^(b)	HLW from LWR Reference Fuel ^(c) Processing
Total g/MTHM	1.46 E6 ^(d)	2.3 E4
U g/MTHM	9.65 E5	4.8 E3
Pu g/MTHM	7.8 E3	1.5 E2
Ci/MTHM		
1 year cooled	2.2 E6	1.6 E6
10 year cooled	3.5 E5	2.5 E5
W/MTHM		
1 year cooled	9.4 E3	8 E3
10 year cooled	8.8 E2	1.3 E3
S.F. n/sec-MTHM		
1 year cooled	9.8 E7	3.6 E9
10 year cooled	5.8 E7	2.5 E9

a. Both fuels at 25,000 MWd/MTHM, 35 MW/MTHM.⁽⁶⁾

b. Fuel fabricated from only enriched uranium.

c. Fuel fabricated from MOX (uranium and plutonium).

d. Includes metallic component of reference fuel.

TABLE 2.19. Radiation from High Level Waste Generated by Processing of Spent
 UO_2 -Bearing LWR Fuel^(a)

Source	Cooling Time					
	160 d	1.4 y	10.4 y	100 y	1000 y	3000 y
Fission Products:						
Ci/MTHM	4.2 E6	1.5 E5	2.4 E5	2.7 E4	1.8 E1	1.7 E1
W/MTHM	1.8 E4	6.4 E3	7.3 E2	7.5 E1	1.44 E-2	1.42 E-2
Actinides:						
Ci/MTHM	8 E3	2.5 E3	7.7 E2	1.5 E2	3.9 E1	1.5 E1
W/MTHM	2.8 E2	7.8 E1	1.9 E1	4.6 E0	1.1 E0	4.0 E-1
n/sec-MTHM ^(b)	1.4 E8	9.2 E7	5.6 E7	2.3 E6	3.1 E5	2.0 E5
Total:						
Ci/MTHM	4.2 E6	1.5 E6	2.4 E5	2.7 E4	5.7 E1	3.2 E1
W/MTHM	1.8 E4	6.4 E3	7.4 E2	8.0 E1	1.1 E0	4.0 E-1

a. Based on 25,000 MWD/MTHM, 35 MW/MTHM, initial U-235 content is 3.3%. The waste contains 0.5% of initial U and Pu, no tritium, Xe, Kr, and 0.1% of I and Br. These percentages are applied to fuel at 160 days after removal from the reactor core.

b. Spontaneous fission neutrons only.

selected radiation data for plutonium separated from LWR-UO₂ fuel is listed in Table 2.21. The 2.2×10^6 n/sec and 1×10^{-1} Ci of gamma radiation, per 5 kg PuO₂, may require modest shielding. Table 2.22 summarizes data for various sources of plutonium.

TABLE 2.20. Actinide Content of High Level Waste from Processing of UO₂ Fuel and of LWR Reference Fuel^(a)

	"Uranium-Only" Fuel Cycle			LWR Reference MOX Fuel Cycle		
	g/MTHM	Cooling Time		g/MTHM	Cooling Time	
		1.4 year (Ci/MTHM)	10.4 year (Ci/MTHM)		1.4 year (Ci/MTHM)	10.4 year (Ci/MTHM)
U	4.8 E3	1.6 E-2	1.4 E-2	4.8 E3	4 E-2	3 E-2
Np	3.3 E2	6.5 E0	6.5 E0	1.5 E2	1.4 E2	1.4 E2
Pu	4.2 E1	3.8 E2	2.6 E2	1.5 E2	1.5 E3	1.1 E3
Am	6.1 E1	1.1 E2	1.1 E2	8.7 E2	7.8 E2	7.8 E2
Cm	6.2 E0	2.0 E3	3.9 E2	2.4 E2	3.8 E4	1.7 E4
Bk	1 E-9	6 E-5	4 E-8	----	----	----
Cf	1 E-7	3 E-6	1 E-6	----	----	----
Total	5.3 E3	2.5 E3	7.7 E2	6.2 E3	4.0 E4	1.9 E4

a. 25,000 MWd/MTHM, 35 MW/MTHM, 0.5% of initial U and Pu, 100% of all other transuranic elements. The percentages apply at 160 days after discharge from reactor core.

2.4.4 Summary

In summary, the generation of various waste streams in the reprocessing of spent reactor fuel has been described. The reference plant receives spent reactor fuel and produces plutonium dioxide and uranium hexafluoride. Various types of waste streams are generated in the reprocessing of the fuel and the purification of the final products. High-level waste is stored in liquid form, onsite, in tanks. Solids are stored in containers or are buried. Intermediate-level liquid wastes are subject to treatment and concentration to a crystalline solid, which is stored in tanks. Low-level liquid wastes are collected and evaporated to a vapor portion that is vented directly to the atmosphere and a concentrated residual liquid that is combined with the stored high-level wastes. The reference reprocessing plant discharges no liquid wastes. The off-gas treatment system scrubs reactive fission products such as iodine from the off-gases. It allows the complete venting to the stack of the noble gases, e.g., krypton, and of tritium and ¹⁴C in the form of water vapor and CO₂.

TABLE 2.21. Properties of Plutonium from LWR-UO ₂ Fuel ^(a)						
	²³⁸ Pu	²³⁹ Pu	²⁴⁰ Pu	²⁴¹ Pu	²⁴² Pu	Total ^(f)
wt %	1.03	65.30	22.49	9.01	2.17	100
αCi/g	17.4	6.15 E-2	0.227	2.58 E-3	3.9 E-3	0.27
βCi/g	---	---	---	1.124 E2	---	10.13
γCi/g ^(b)	2 E-4	4.7 E-6	---	2 E-4	---	2.5 E-5
n/sec-g ^(c)	2.8 E3	2.2 E-2	1.02 E3	2.3 E-2	1.73 E3	3 E2
n/sec-g ^(d)	1.4 E4	45	1.7 E2	~10	~10	2 E2
Total n/sec-g	1.7 E4	45	1.2 E3	~10	1.7 E3	5 E2
W/g ^(e)	0.577	1.91 E-3	7.07 E-3	α7.99 E-5 β4.64 E-3	1.15 E-4	9.2 E-3

a. 25,000 MWD/MTHM, 35 MW/MTHM, 7.826 kg Pu discharged/MTHM. Initial ²³⁵U = 3.3%.

b. At E >100 kev, α decay gamma only.

c. Spontaneous fission.

d. (α,n) on oxygen in PuO₂.

e. Based on Q_α and Q_β.

f. Totals are sums of isotopic contribution in relation to their wt % of elemental plutonium.

TABLE 2.22. Isotopic Content and Radiations from LWR Plutonium

	Charge to Reference LWR ^(a)	LWR-U Discharge ^(b)	LWR-Pu Charge ^(c)	LWR-Pu Discharge ^(c)
Pu-238, wt %	2.44	1.03	2.69	3.04
Pu-239, wt %	47.02	65.03	44.57	37.75
Pu-240, wt %	26.04	22.49	25.70	27.94
Pu-241, wt %	15.51	9.01	16.67	18.90
Pu-242, wt %	8.99	2.17	10.38	12.53
α Ci/gPu ^(d)	0.51	0.27	0.55	0.62
β Ci/gPu ^(e)	17.43	10.13	18.74	21.24
γ Ci/gPu ^(f)	38 E-6	25 E-6	41 E-6	46 E-6
W/gPu ^(g)	17.5 E-3	9.2 E-3	19 E-3	21.1 E-3
n/sec-gPu ^(h)	4.9 E2	3.0 E2	5.2 E2	5.9 E2
kg/MTHM	15.0	7.826	49.8	35.5

a. Based on 30 wt % LWR fuel is MOX at ~5 wt % PuO₂ in UO₂, 70 wt % LWR fuel is 3.3% enriched UO₂. Assumed to be the equilibrium composition, i.e., reprocessing plant² product and MOX fabrication plant feed.

b. Based on 3.3% enriched feed, 25,000 MWd/MTHM and 35 MW/MTHM irradiation.

c. From Reference 6.

d. Includes no contribution of Am.

e. Includes only Pu-241.

f. For E >100 kev. Includes only α or β decay gammas.

g. Based on Q _{α} and 1/3 Q _{β} .

h. From spontaneous fission only. (α ,n) on oxygen about doubles the emission rate.

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2.5 WASTES FROM MIXED OXIDE FUEL FABRICATION PLANTS

2.5.1 Process Description

Plutonium from the reprocessing plant is combined with natural uranium in the fabrication plant to produce mixed oxide LWR fuel (MOX). Plutonium is shipped to the plant as the dry oxide (see Table 2.23 for radiation properties) in sealed, overpacked containers (see Section 22). The sealed containers are put into the glove boxes that are continually isolated from the plant atmosphere and the environment. Batches of uranium dioxide are transferred in similar fashion. Within the glove boxes, the following processes are performed.

1. blending of PuO_2 and UO_2 powders
2. cake-pressing of mixed-oxide-blend (MOX) powder
3. granulation and reblending of caked MOX
4. forming MOX granules into "green" pellets
5. high-temperature sintering of green pellets
6. finish grinding of sintered pellets
7. cleaning and inspection of finished pellets
8. loading finished pellets into fuel cladding tubes
9. fuel cladding decontamination and seal-welding.

At this point, the plutonium in the fuel is contained in sealed metal tubes that are outside the glove box enclosures. Subsequently, the fuel elements are subjected to the following additional routines by direct contact.

1. degreasing
2. fuel cladding etching and rinse
3. x-ray inspection of fuel pellet arrangement
4. autoclaving of fuel elements
5. fuel cladding leak-check and dimensional inspection
6. packaging, storage, and shipment.

The contaminated waste materials generated in these two operational areas range in transuranic (TRU), primarily plutonium, concentration from gross amounts to TRU-suspect concentrations. The latter wastes are classified as such by proposed regulations in 10CFR20 because they come from restricted areas of a TRU facility.⁽¹⁾ Table 2.24 shows a summary of characteristics of the waste streams. These wastes are identified and arranged by type in the following sections.

2.5.2 Transuranic-Contaminated Combustible Solid Waste

Combustible solids considered to be contaminated with transuranic elements are generated in two specific areas of a MOX fabrication plant (Figure 2.26). The first is in the operating area defined as restricted but not exposed to glove box interior atmospheres and/or effluents.⁽⁴⁾ Combustible materials⁽⁵⁾

TABLE 2.23. Mixed Oxide Fabrication Plant Plutonium Feed ^(a)						
	²³⁸ Pu	²³⁹ Pu	²⁴⁰ Pu	²⁴¹ Pu	²⁴² Pu	Total
wt %	2.44	47.02	26.04	15.51	8.99	100
αCi/g ^(b)	17.4	6.15 E-2	0.227	2.58 E-3	3.9 E-3	0.51
βCi/g ^(c)	---	---	---	1.124 E2	---	17.43
γCi/g ^(d)	2 E-4	4.7 E-6	---	2 E-4	---	38 E-6
n/sec-g ^(e)	1.68 E4	45.02	1.19 E3	~10	1.74 E3	9 E2
W/g ^(f)	0.577	1.91 E-3	7.07 E-7	4.64 E-3 ^(g)	1.15 E-4	17.5 E-3

a. Derived from data of Reference 2 and based on discharge from the reference LWR with 30 wt % MOX fuel and 70 wt % 3.3% enriched UO₂ fuel, 25,000 MWd/MTHM, 35 MW/MTHM, and an approximate self generation ratio ²(SGR) ~1.0.

b. No Am included.

c. Only ²⁴¹Pu included.

d. at E >100 kev, decay gammas.

e. Spontaneous fission and (α,n) on oxygen in PuO₂.

f. Based on Q_α and 1/3 Q_β.

g. Beta power only.

TABLE 2.24. Amounts and Activities of Plutonium-Contaminated Waste from Mixed-Oxide Fuel Fabrication

	Combustible Solid ^(a)	Non- Combustible Solid ^(a)	HEPA Filters ^(a)	Liquid ^(b)
Waste Volumes				
m ³ / MTHM in MOX	4.1	0.8	0.3	0.06
m ³ / 13.7 MTHM in MOX ^(c)	56	11	4	0.8
m ³ / 450 MTHM in MOX ^(d)	1.9 E3	360	140	27
Plutonium Mass				
g/m ³ of waste	30	28	130	750
g/MTHM in MOX	120	22	39	45
g/13.7 MTHM in MOX	1.7 E3	300	530	620
g/450 MTHM in MOX	5.7 E4	E4	1.7 E4	2 E4
Pu Alpha Activity ^(e)				
Ci/m ³ of waste	15	14	70	380
Ci/MTHM in MOX	60	11	20	23
Ci/13.7 MTHM in MOX	850	150	260	310
Ci/450 MTHM in MOX	2.9 E4	5 E3	9 E3	E4
Pu Alpha-Beta Activity ^(e)				
Ci/m ³ of waste	540	500	2.3 E3	E4
Ci/MTHM in MOX	2.2 E3	400	700	810
Ci/13.7 MTHM in MOX	3.1 E4	5.4 E3	9.4 E3	E4
Ci/450 MTHM in MOX	E6	1.8 E5	3.2 E5	4 E5

- a. Values based on simple average of predictions found in References 3 and 4.
- b. Calculated from Reference 3.
- c. Represents MOX fraction of fuel for a GWe-year; 30% of a total 45.6 MTHM in enriched-U plus MOX fuels.
- d. Represents MOX fraction of fuel for the 1500 MTHM-year case; 30% of a total 1500 MTHM in enriched-U plus MOX fuels.
- e. Data based on "Reference" Pu activity of 0.5 Ci/g alpha and 18 Ci/g alpha plus beta.

a. Data on m³ and g Pu/MTHM from Table 2.24, ratios for waste categories from Reference 3, and ratio of glove box/operating-area waste amounts and Pu content from Reference 4.

FIGURE 2.26. Plutonium Combustible Solid Waste from the Mixed-Oxide Fuel Fabrication Plant^(a)

include neoprene and latex, polyvinyl chloride (PVC) and polyethylene plastics,* various types of cloth, and various types of paper (cellulosics). It has been estimated that this category of materials could have a transuranic activity of 7 μCi (alpha) or 0.3 mCi (alpha, beta) per gram of waste (14 μg Pu/g waste).^(3,4)

Combustible materials that are exposed to glove box atmospheres and/or effluents comprise the second category of waste. This waste could also include neoprene, latex, PVC, polyethylene, various types of cloth, and cellulosics. The TRU content of this waste type is estimated to be 0.2 mCi (alpha) or 5.4 mCi (alpha, beta) per gram of waste (0.3 mg Pu/g waste).^(3,4)

It has been estimated that a 1500 MTHM-year Pu-recycle rate (450 MT of MOX, 22.4 MT of Pu) may result in the generation of $2 \times 10^3 \text{ m}^3$ ($4 \text{ m}^3/\text{MTHM}$) of combustible solid waste from mixed oxide fuel fabrication.^(4,6) On a 1000 MWe-year basis, $\sim 60 \text{ m}^3$ of this type waste may be expected from the fabrication of 13.7 MT of MOX (30% of 45.6 MTHM fuel). An estimated 15 Ci (alpha) or 540 Ci (alpha, beta)/ m^3 can be expected in TRU combustible solid wastes.** Thus, on a 1000-MWe-year basis, 850 Ci (alpha) or $1.7 \times 10^3 \text{ g}$ of Pu might appear with these wastes. This would represent a 0.25% plutonium loss.

2.5.3 Transuranic Noncombustible Solid Waste

Noncombustible wastes contaminated with transuranic elements typically also come from both plant areas (Figure 2.27). Noncombustibles that are not exposed to the glove box atmospheres or effluents are metal, glass, and mineral insulation material.^(4,5) It is estimated that the transuranic activity of this category of waste is in the same magnitude as the combustible waste from this area, e.g., 1.5 μCi (alpha) or 54 μCi (alpha, beta) per gram of waste (3 μg Pu/g waste).^(3,4)

Noncombustibles that have been exposed to glove box atmospheres or effluents include metal transfer containers, metallic equipment, metal ductwork, and reject fuel-clad tubing.^(4,5) The estimated transuranic contamination levels approximate those for in-box combustibles, e.g., 0.1 mCi (alpha) or 3.6 mCi (alpha, beta) per gram of waste (0.2 mg Pu/g waste).^(3,4)

Another noncombustible in-box waste is scrap or residual PuO_2 or $(\text{U}, \text{Pu})\text{O}_2$. This material, which is likely to be recycled, could have TRU activities of $\sim 0.5 \text{ Ci}$ (alpha, beta)/g of the mixed oxides to $\sim 12 \text{ Ci}$ (alpha, beta)/g of PuO_2 ,⁽³⁾ an estimated potential of 2% Pu loss.⁽⁸⁾

*Polyethylene and PVC plastics have been included with combustible solids despite their poor combustion characteristics. The quantities of this type waste are estimated to be 68 kg/MTHM of out-of-box and 950 kg/MTHM of in-box combustible wastes, based on a survey which showed plastics to be 20% of out-of-box and 36% of in-box total solid waste.⁽⁵⁾

**Estimates from other reported data include: 10 Ci (alpha)/ m^3 (Reference 3); 8 Ci (alpha)/ m^3 (Reference 7); and 300 Ci/ m^3 (apparent Pu specific activity of 6.8 Ci/g) (Reference 4).

a. Data on m³ and g Pu/MTHM from Table 2.24, ratios for waste categories from Reference 3, and ratio of glove box/operating-area waste amounts and Pu content from Reference 4.

FIGURE 2.27. Plutonium Noncombustible Solid Waste from the Mixed-Oxide Fuel Fabrication Plant^(a)

An important fraction of transuranics will be found in discarded high-efficiency particulate air (HEPA) filters used in glove box ventilation systems. This noncombustible solid waste will be described separately in a subsequent section.

The estimated generation rate for noncombustible solids, excepting fuel scrap and HEPA filters, is $360 \text{ m}^3/1500 \text{ MTHM-year}$, or $11 \text{ m}^3/1000 \text{ MWe-year}$. For the 1000 MWe-year, the waste would be contaminated with 160 Ci (alpha) or $\sim 310 \text{ g}$ of plutonium, representing a 0.05% Pu loss.

2.5.4 Transuranic Liquid Waste

Although the MOX fabrication process is essentially a dry operation, four steps do require a liquid for cleaning and decontamination.^(4,8) Figure 2.28 identifies these liquid streams. After finish grinding, MOX fuel pellets are washed with water. Due to the insolubility of transuranic oxides, contamination would result only from the suspension of fine particles in the wash water.

After pellets are loaded in fuel-cladding tubes, the resulting elements are subjected to decontamination (before or after final end-cap welding) and a subsequent degreasing procedure. Both the decontamination and degreasing fluids are suspect for TRU content.⁽¹⁾

Prior to autoclaving, the elements are subjected to etching. Generated in this step are HNO_3 -HF etching solutions, aluminum nitrate stop-etch solutions, and water used as a rinse, all TRU-suspect streams.

The generation rate for TRU-contaminated liquids for the 1500 MTHM Pu-recycle basis has been estimated to be 2.7×10^4 liters. For the 1000-MWe-year, 800 liters of liquid waste could be expected. Using an estimate of 0.4 Ci (alpha) or 14 Ci (alpha, beta)/liter, TRU liquids might appear with activities of 315 Ci (alpha) or 10^4 Ci (alpha, beta) per 1000 MWe-year.⁽³⁾ This would amount to about 620 g Pu/1000 MWe-year, or a plutonium loss of 0.09.

Adding a scrap recovery system to the plant flow sheet, recycling scrap PuO_2 and $(\text{U}, \text{Pu})\text{O}_2$, would generate another TRU-contaminated liquid stream.⁽⁶⁾ This liquid would appear in the form of an acidic raffinate from a solvent-extraction step. Extensive recycling to reduce plutonium losses is likely to be practiced.

The reference fabrication plant uses evaporation to eliminate liquid wastes. The resulting solid residue is dried and would be removed as a noncombustible TRU waste, or would be recycled to aqueous recovery and purification operations. Hence the amounts of TRU (plutonium) waste shipped from the fabrication plant will depend to some extent on the specific recycle and recovery operations practiced.

2.91

a. Based on data from Reference 3.

FIGURE 2.28. Plutonium Liquid Waste from the Mixed-Oxide Fuel Fabrication Plant^(a)

2.5.5 HEPA Filter Waste (Figure 2.29)

The mixed oxide fabrication processes within the glove boxes will generate particles of PuO_2 and $(\text{U}, \text{Pu})\text{O}_2$ powder. A portion of these particles will become entrained in the ventilation flow. High-efficiency air (HEPA) filters are used to clean this air stream before discharge to the atmosphere. Hence filter media become contaminated with TRU material, and, when replaced, HEPA filters are a relatively high-volume solid waste at a MOX fabrication plant. Estimated data for a comparable MOX production facility set an average transuranic concentration in spent HEPA filters at 125 Ci (alpha) or 4.5×10^3 Ci (alpha, beta)/ m^3 .⁽³⁾

Filters in the ventilation exhausts from operational areas outside glove box enclosures also will constitute TRU-contaminated solid waste. Transuranic concentration in these HEPA filters is expected to be much less than that for glove box ventilation filters.

For the 1500 MTHM-year fabrication basis, it is estimated that HEPA filter wastes are generated at a rate of $140 \text{ m}^3/\text{year}$, or $4 \text{ m}^3/1000 \text{ MWe-year}$.^(3,4) In this waste, transuranic contamination would amount to 9×10^3 Ci (alpha)/1500 MTHM-year or 260 Ci (alpha)/1000 MWe-year. For the 1000 MWe-year, these activities represent 520 g Pu, or a plutonium loss of 0.08%.

a. Data on m³ and g Pu/MTHM from Table 2.24, ratios for waste categories from Reference 3, and ratio of glove box/operating-area waste amounts and Pu content from Reference 4.

FIGURE 2.29. Plutonium Ventilation-Filter Waste from the Mixed-Oxide Fuel Fabrication Plant^(a)

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2.6 WASTES FROM DECOMMISSIONING OF FUEL CYCLE FACILITIES

Materials resulting from the decontamination and decommissioning (D&D) of fuel cycle facilities must be included in any complete waste management scheme for the fuel cycle. In addition, D&D is receiving increased attention because of its importance to arriving at a complete assessment of nuclear power economics. D&D operations and results are detailed in Section 15 of this report. The characterization of materials emerging from decommissioning, i.e., the identity of materials, their volume, and the contamination levels associated with them, are the concern of the present section.

The commercial fuel cycle facilities expected to contribute to D&D wastes are the fuel reprocessing plants, mixed-oxide fuel fabrication plants, reactors, fuel storage basins, and facilities for interim storage of trans-uranium wastes. Although experience obtained in the D&D of first-generation reactors⁽¹⁾ and a number of contaminated ERDA facilities⁽¹⁾ indicates that generic techniques are available, experience on D&D of facilities representative of the commercial fuel cycle is not available. The recent conference on D&D⁽¹⁾ summarized present experience.

Three modes of decommissioning are presently considered rather generally by the nuclear industry, although a consistent nomenclature is apparently not in common use. These three modes are described in Section 15 of this report. The choice of mode and decontamination operations might be made on the basis of such factors as the type of facility, the type and level of contamination, location of the facility, and the cost/risk ratio of applying the alternative modes to the particular facility. All modes involve decontamination by treatment with liquids to remove radioactivity from surfaces, with the resultant generation of contaminated liquids requiring additional waste treatment (e.g., volume reduction) operations. The mode that most requires planning for subsequent waste management, i.e., dismantling, is the one that involves shipping of residual materials to storage.

Since experience in D&D of modern commercial fuel cycle facilities is not available, conceptual studies of decommissioning of reference plants are the only bases at present for discussion of materials type, volume, and associated contamination levels. Several such conceptual studies are in progress, and the results are expected to be published in 1976. One study of the D&D of LWRs is expected to be published soon;⁽²⁾ the only prepublication data available are presented in preliminary form in Table 2.25. An independent study of the decommissioning of LWRs is also expected to be initiated soon.⁽³⁾ Some factors involved in the decommissioning of facilities for interim storage of transuranium wastes are discussed in Section 20 of this report, but quantitative estimates are not available at this time. A

TABLE 2.25. Volumes and Contamination
Categories of Residues
from Decommissioning of
Light Water Reactors^(a,b)
Volume (m³)^(c)

Reactor Type	Non- radioactive Materials	Radioactive Materials
BWR	36,700	23,900
PWR	29,800	7,000

- a. 1000 MWe capacity.
- b. Reference 2, data to be published in a study of decommissioning by the Atomic Industrial Forum.
- c. Includes plant components, building materials, and a relatively small contribution from decommissioning wastes probably in the form of evaporator bottoms.

discussion of the criteria for decommissioning of a mixed oxide facility has been presented,⁽⁴⁾ and the decommissioning of a fuel reprocessing plant has been discussed in terms of standards.^(5,6) Results of a conceptual study of the decommissioning of a mixed oxide fuel fabrication plant and a fuel reprocessing plant are also expected to be available in 1976;⁽³⁾ prepublication results are summarized in Table 2.26 and Table 2.27.

In summary, the waste materials emerging from D&D operations are expected to fall into the general categories of metallic process components, building materials, and decontamination solutions or the residues from volume reduction operations on the solutions. Although few quantitative results are presently available, the area of D&D is beginning to receive attention, and additional quantitative estimates are expected soon.

TABLE 2.26. Contaminated Waste from Dismantling a 1500 MTU/year Fuel Reprocessing Facility^(a)

	Process Building and Waste Tank Galleries	Liquid Waste Storage Tanks
Equipment		
Volume (m ³)	2,000	1100
Weight (MT)	1,600	910
Avg fission product contamination (Ci/m ³)	0.04	1.4
% >10 nCi/g TRU ^(b)	20	0
Stainless steel liners		
Volume (m ³)	31	130
Weight (MT)	245	1000
Avg fission product contamination (Ci/m ³)	3	3.2
% >10 nCi/g TRU ^(b)	7	0
Concrete		
Volume (m ³)	594	--
Weight (MT)	1,400	--
Avg fission product contamination (Ci/m ³)	14	--
% >10 nCi/g TRU ^(b)	43	--
Total shipping volume	6,510 m ³ ; 1,090 m ³ = >10 nCi/g TRU	

a. Reference 3.

b. Nanocuries per gram of transuranic waste, essentially plutonium.

TABLE 2.27. Contaminated Waste from Dismantling a 300 MTHM/year Mixed-Oxide Fabrication Facility^(a)

	Volume (m ³)	Weight (MT)
Equipment	1,500	1,300
Structural material	350	1,300
Miscellaneous	550	300
Total	2,400	2,900
% >10 nCi/g TRU ^(b)	62.7	
Total shipping volume 4,000 m ³ ; 2,600 m ³ = >10 nCi/g TRU		

a. Reference 3.

b. Transuranic waste-essentially plutonium.

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3.0 PROJECTIONS OF LWR FUEL CYCLE QUANTITIES

3.0 PROJECTIONS OF LWR FUEL CYCLE QUANTITIES

3.1 BASIC ASSUMPTIONS

Projections of waste quantities and associated management activities for the LWR fuel cycle are presented here to indicate the future requirements for waste management facilities and services to be provided by Federal and commercial organizations.

Quantities of, characteristics of, and management requirements for wastes from the LWR fuel cycle will depend on total energy production and growth, the penetration of total energy production by electrical energy, and the fraction of electrical energy produced by light water reactors. The United States Energy Research and Development Administration Office of Planning and Analysis periodically prepares forecasts, based on extrapolation of recent trends in operation, construction, and planning of nuclear power plants, in the form of sets of probable conditions or reference cases.⁽¹⁾ Uncertainties and trends in the demand for power in general and in the conditions affecting nuclear power growth in particular have recently necessitated frequent revision of projections. Thus the projections presented here differ somewhat from those in a detailed study issued in 1974.⁽²⁾ Moreover, the most probable reference case has been revised from a moderately high growth case in early 1975 to a moderately low growth case in late 1975. This latter case is the basis for projections presented here.

3.1.1 Total Electric and Nuclear Capacities

The power-growth case used here assumes a moderately low growth of electrical energy, the delay of spent fuel reprocessing until late calendar year 1978, and the delay of plutonium recycle until calendar year 1981. The kilowatt-hour production in this case grows at 6.0% per annum through 1985 and then declines to a 5.4% per annum growth through the year 2000. Table 3.1 lists projections of total electrical generating capacity and total installed nuclear capacity by year.

3.1.2 Bases for Projections of Electric Operating Capacity, Materials Flows, and Waste Generation Involving LWRs

Projections of parameters have been selected to characterize the LWR fuel cycle in the period 1975 through 2000. The assumptions upon which these projections are based are collected in this section. Tables of projections are presented in Section 3.2. These parameters include LWR operating capacity (Table 3.2), the flow and accumulation of primary wastes, and the flow of various other materials necessary to the operation of a complete LWR fuel cycle.

TABLE 3.1. Characteristics of the Mod-Low Energy Growth Case

Year	Total Electric Generating Capacity,	Total Installed Nuclear Capacity, ^(a)
	GWe	GWe
1973	436	18.4
1980	620	76.0
1985	800	185.0
1990	1040	340.0
2000	1750	800.0

a. Includes all types of reactors. See Table 3.2 for projections of operating capacity for LWRs only.

Assumptions basic to the projections have been previously described,⁽²⁾ and are briefly summarized here. The ratio of PWRs to BWRs = 2:1. Plant factors are 40% during the pre-operational period between fuel loading and commercial operation, 65% during the first two years of commercial operation, 70% for years 3 through 15, and following the 15th year, decline at 2 percentage points per year to a minimum of 40%. Uranium is recycled as soon as material can be returned through normal fuel cycle timing. Plutonium is recycled beginning with 25% of the reactors in 1981, 50% of reactors in 1982, 75% of reactors in 1983, and without constraint in following years except that sufficient plutonium is withheld for use in FBRs.

The burnup of fuel at discharge determines several factors of importance to waste management, including the composition, thermal power, and gamma power of high-level waste. Average burnup at discharge and power level during irradiation will differ slightly for BWRs and PWRs and will increase somewhat during the period 1975 to 2000. Much of the planning for an LWR fuel cycle has been done on the basis of a fuel burnup of 33,000 MWd/MTHM. This burnup level is expected to be characteristic of future LWR fuel, but the average burnup will be lower for fuel discharged to the present, now in storage, and slated for processing between 1975 and 2000. With this expectation, data on isotopic composition, thermal power, and gamma power associated with high-level waste and with cladding hulls presented in Sections 2.3 and 2.4 are based on a burnup of 25,000 MWd/MTHM and a specific power of 35 MW/MTHM. Calculations of projections, however, are presently available only for a burnup of 33,000 MWd/MTHM, and these numbers are the basis for the projections presented in Section 3.2. Since higher burnup will increase the amounts of longer-lived radioisotopes to be found in waste, these projections will be conservative in that estimates of activities associated with the wastes will tend to be high.

Plutonium recycle in the PWR assumes self-sustaining recycle that uses a mixture of assemblies containing either enriched uranium (PWR-U) or plutonium mixed with natural uranium (PWR-Pu). These assemblies are assumed to be shipped and reprocessed together, such that the plutonium product, and wastes, will be blended. At steady state: 1) 1.2 atoms of fissile plutonium replace 1.00 atom of ^{235}U in the plutonium-fueled assemblies; 2) 32.4% of the assemblies in the reactor are fueled with plutonium; and 3) the plutonium used in the plutonium-fueled assemblies has an initial fissile content of 61.3% and a fissile content at discharge of 55.4%. The plutonium produced in the enriched uranium elements (70.64% fissile) is blended with the product of the plutonium elements (55.4% fissile) to prepare the plutonium for the next refueling (61.3% fissile). LWR fuel is assumed to be reprocessed after a postirradiation decay time of 150 days.

The accumulations of ore tailings from milling uranium ores (Table 3.3) are based on the assumption that 95% of the uranium is extracted from ores that assay 0.2% uranium, and that the unrecovered 5% and all radioactive daughters remain with the tailings. The volume of tailings is estimated on the basis of 0.264 m^3 being formed per kilogram of recovered U_3O_8 . The requirements for mining of uranium are based upon a tails assay in uranium enrichment of 0.2% ^{235}U and allows for the recycle for reenrichment of uranium that contains greater than 0.4% ^{235}U .

The annual production of fissile materials (Table 3.4) is defined as the annual quantity stockpiled following reprocessing. The quantities in the columns labeled "used" are the annual quantities withdrawn from this stockpile.

The projected annual loads of fuel fabrication plants (Table 3.5) includes total heavy metal. The shipment load for plutonium recycle in light water reactors (Tables 3.6, 3.7), on the other hand, includes only those portions of the fuel assemblies that are fueled with mixed oxide.

The projected fuel reprocessing schedule (Tables 3.8-3.10) differs slightly from the moderately-low growth case in that reprocessing begins on a small scale in 1977 rather than in 1978.

High-level wastes (Tables 3.11-3.13) are those aqueous raffinates resulting from the operation of the first cycle solvent extraction system, or equivalent, and the concentrated wastes from subsequent extraction cycles, in a facility for reprocessing irradiated reactor fuels. These wastes contain virtually all of the nonvolatile fission products, several tenths wt % of uranium and plutonium originally in the spent fuels, and all the other actinides formed by transmutation of the uranium and plutonium in the reactors. Present regulations call for these wastes to be solidified within 5 years after they are generated and for the resultant stable solids to be shipped to a federal repository within 10 years after the liquids are generated.⁽³⁾

3.4

In the characterization of these wastes, reprocessing flow sheets were assumed that minimized, wherever possible, additions of chemicals which might be particularly troublesome in subsequent waste management operations, specifically, sodium and sulfate ions. Product losses are taken as 0.05 wt% of the Pu to the LWR cladding; 0.1 wt% of the Pu and U to the aqueous raffinate;* and 0.01 wt% of the Pu and U to the solvent during stripping (which ultimately appears in the solvent regenerants).

The actual volumes of high-level waste will depend on many factors that must be resolved through actual operating experience. Although waste management economics will tend to force the final volumes toward smaller values, additional wastes arising from maloperations, equipment decontamination, etc., will tend to increase them. Similarly, possible future requirements for higher quality waste products (less leachable in water, more radiation-resistant) may also result in higher volumes of waste per ton of fuel processed. In view of these possibilities, an upper limit volume for solidified high-level waste of 0.06 m^3 per MTHM processed (based on waste in the form of oxides with minor additions of NaOH or B_2O_3 to prevent volatilizing of some components) is used in calculating volumes of solidified high-level waste.

Although solidification of commercial high-level waste has not as yet been practiced, packaging concepts include the use of steel canisters $\sim 0.3 \text{ m}$ in diam by $\sim 3 \text{ m}$ long. As many as 12 such canisters (each containing 0.2 m^3 of 10-year-old solidified waste) could be shipped by rail in a single iron- or lead-shielded cask.^(4,5)

Projections on cladding wastes (Table 3.14) are based on volumes of the residual Zircaloy and stainless steel cladding and structural components of the fuel assemblies that remain after the fuel cores have been dissolved. Many of the assumptions used to construct projections are based on an earlier study of cladding wastes.⁽⁶⁾ In addition to neutron-induced isotopes, their main source of radioactivity, the cladding is assumed to contain 0.05 wt% of the actinides and 0.05 wt% of the fission products originally in the spent fuel. Although cladding may be compacted for storage, the volumes are presented here on an uncompacted basis.

Shipments and accumulations of fission product tritium (Table 3.15), 1 year after its separation from spent fuel at reprocessing plants, are based on the assumptions that this tritium is recovered in water at a concentration of 200 Ci/liter and is then immobilized as $\text{Ca}(\text{OH})_2$, which is shipped, probably by motor freight, to a repository for long-term storage. Most of the fission product tritium contained in spent fuel will accumulate in the low-level

*This level of residual contamination is adequate for projections, similar to the 33,000 MWd/MT burnup. Calculations for Section 2.4 assumed 0.5 wt% U and 0.5 wt% Pu in high-level waste.

aqueous waste of the fuel reprocessing plant (Table 2.10). Within present regulations, this tritium can be released to the environment by discharging the low-level waste to surface streams, depending on site characteristics, or by vaporizing it into the atmosphere. However, in the future, the separation of tritium from spent fuel may be required, either before fuel dissolution or from low-level aqueous waste after the normal sequence of Purex operations. The necessity and feasibility of separating and storing tritium is currently being debated. Although a number of separation schemes have been examined, and some have received some development work, technology for collection is not ready for commercial application. The projected accumulations of fission-product tritium presented here, therefore, illustrate a set of possible options rather than current practice.

Projections on the accumulation of fission product iodine (Table 3.16) are based on the assumption that 99.9% of the iodine is recovered at the reprocessing plants, converted to KI, also as an example rather than an indication of most probable practice, and stored for 1 year before shipment to a repository or to a disposal site.

Current regulations permit the release of noble gases at the fuel reprocessing plant, but a requirement that noble gases be collected in the future and stored is currently being debated. The basic collection technology is available, but the feasibility of quantitative collection and storage, on the scale expected for commercial fuel reprocessing, has not been demonstrated. Therefore, the projections presented here (Table 3.17) are only an illustration of the option of collecting and storing the noble-gas fission products. For the projected accumulation of noble-gas fission products, it is assumed that the mixed krypton and xenon isotopes are packaged in standard 50-liter gas cylinders at a pressure of 156 kg/cm², stored for 1 year at the reprocessing plant, and then shipped by rail to a repository for long-term storage. An individual shipment is assumed to consist of six gas cylinders.

Carbon-14 is produced in LWR fuel by the action of neutrons on ¹⁴N, a constituent of the nitrogen impurity in the oxide fuel. The projections listed in Table 3.18 are based on the assumption that a nitrogen impurity of 25 ppm is present in oxide fuels. As in the cases of tritium and noble gases, the questions of necessity and feasibility of collecting and storing ¹⁴C released from spent fuel at the reprocessing plant have not been resolved. The projections of accumulated ¹⁴C, therefore, represent only the results of a possible option in its management.

Projections of beta-gamma solid wastes are based on the generation of 1100, 2300, and 450 m³ per metric ton of plutonium processed at fuel preparation, fabrication, and reprocessing plants, respectively. In addition, the accumulations include 113 m³/year of waste from each installed 1000-MW LWR.

For the projections of Table 3.19, no allowance is made for their volume reduction by compaction or other methods. The volumes are the sums of combustible and noncombustible wastes.

Table 3.20 presents projections of plutonium alpha (TRU, low β - γ) wastes assuming they are shipped and accumulated at a repository 1 year following their generation. Volumes are based on the generation of 283, 566, and 113 m³ of waste per ton of plutonium processed in fuel preparation, fabrication, and reprocessing plants, respectively. The average concentration of plutonium (1% ²³⁸Pu, 59% ²³⁹Pu, 24% ²⁴⁰Pu, 12% ²⁴¹Pu, 4% ²⁴²Pu) in the wastes is taken as 9 g/m³ before treatment for volume reduction. The volumes listed are the sums of combustible and noncombustible wastes.

Accumulations of alpha-beta-gamma (TRU, intermediate-level β - γ) solid wastes (Table 3.21) were constructed on the basis of 283 m³ of such waste being generated per metric ton of plutonium recovered from spent fuels at reprocessing plants and it is also assumed that 0.025% of the fission products in the spent fuels is present. The volumes listed are the sums of combustible and noncombustible wastes.

3.2 PROJECTIONS

The projections are presented in Tables 3.2 through 3.21 as annual amounts and also, for some factors, as accumulated amounts for succeeding years from 1975 through 2000. Tabulations for some factors cover a period later than 1975-2000 because of the delayed initiation of the operation from which they are derived, e.g., fuel reprocessing, plutonium recycle, and high-level waste solidification.

TABLE 3.2. Projected Nuclear LWR Operating Capacity^(a)

Year	GWe
1975	38.0
1976	44.3
1977	51.3
1978	58.4
1979	65.5
1980	75.5
1981	92.3
1982	112.7
1983	133.9
1984	157.4
1985	179.4
1986	202.0
1987	225.6
1988	254.0
1989	284.0
1990	315.6
1991	348.7
1992	382.7
1993	416.6
1994	450.7
1995	483.9
1996	516.2
1997	547.6
1998	579.1
1999	609.7
2000	638.6

a. Reference 7, based on assumptions summarized in Section 3.1.2.

TABLE 3.3. Projected Accumulation of Uranium Ore Tailings^(a)

Year	Annual Addition				Accumulation Through End of Year			
	Volume (Millions of m ³)	Heavy Element (MT)	Radio- activity (MCi)	Thermal Power (kW)	Volume (Billions of m ³)	Heavy Element (MT)	Radio- activity (MCi)	Thermal Power (kW)
1975	2.77	448	0.03	0.71	0.012	1893	0.1	2.9
1976	3.15	510	0.04	0.80	0.015	2403	0.2	3.7
1977	3.50	566	0.04	0.89	0.018	2969	0.2	4.6
1978	3.77	609	0.04	0.96	0.022	3578	0.3	5.6
1979	4.54	734	0.05	1.16	0.027	4312	0.3	6.7
1980	5.77	933	0.07	1.47	0.032	5246	0.4	8.2
1981	6.78	1096	0.08	1.73	0.039	6341	0.4	9.9
1982	7.88	1274	0.09	2.01	0.047	7616	0.5	11.8
1983	9.20	1487	0.11	2.34	0.056	9102	0.6	14.1
1984	9.77	1579	0.11	2.49	0.066	10681	0.8	16.6
1985	10.78	1743	0.13	2.75	0.077	12424	0.9	19.3
1986	11.72	1895	0.14	2.99	0.089	14319	1.0	22.2
1987	13.36	2160	0.16	3.41	0.102	16479	1.2	25.6
1988	14.58	2357	0.17	3.72	0.117	18836	1.3	29.3
1989	16.03	2590	0.19	4.08	0.133	21426	1.5	33.3
1990	17.51	2830	0.20	4.46	0.150	24256	1.7	37.7
1991	18.87	3050	0.22	4.81	0.169	27306	1.9	42.4
1992	20.14	3255	0.24	5.13	0.189	30561	2.1	47.5
1993	21.63	3495	0.25	5.51	0.211	34056	2.4	52.9
1994	23.06	3726	0.27	5.88	0.234	37782	2.7	58.7
1995	24.39	3943	0.28	6.22	0.258	41725	2.9	64.8
1996	25.81	4171	0.30	6.58	0.284	45896	3.2	71.2
1997	27.36	4422	0.32	6.97	0.311	50318	3.5	78.1
1998	28.46	4600	0.33	7.25	0.340	54918	3.9	85.2
1999	29.84	4823	0.35	7.60	0.370	59740	4.2	92.7
2000	30.86	4987	0.36	7.86	0.400	64727	4.5	100.4

a. Reference 7, based on assumptions summarized in Section 3.1.2.

TABLE 3.4. Materials Consumed, Produced, and Stockpiled for LWR Power (metric tons)^(a)

Year	Uranium Ore Mined	Fissile Plutonium		
		Prod	Used	Net
1975	8941	2.65	0.0	7.13
1976	10171	3.95	0.0	11.09
1977	11296	5.31	0.0	16.39
1978	12155	6.46	0.0	22.85
1979	14660	7.72	0.0	30.57
1980	18630	8.88	4.40	35.05
1981	21871	10.08	12.90	32.23
1982	25430	13.11	18.90	26.43
1983	29673	18.18	14.30	30.32
1984	31522	23.02	16.20	37.14
1985	34790	24.69	20.90	40.93
1986	37815	28.92	22.40	47.45
1987	43107	34.20	28.30	53.35
1988	47048	38.41	34.20	57.56
1989	51701	44.29	38.20	63.65
1990	56492	50.52	41.10	73.07
1991	60876	56.44	43.40	86.12
1992	64971	62.26	48.70	99.68
1993	69763	68.16	51.50	116.34
1994	74377	75.25	51.00	140.59
1995	78694	81.60	49.40	172.79
1996	83255	86.93	45.50	214.22
1997	88260	91.54	40.50	265.27
1998	91814	95.35	36.10	324.52
1999	96260	98.61	38.60	384.53
2000	99538	102.08	31.60	455.01

a. Reference 7, based on assumptions summarized in Section 3.1.2.

TABLE 3.5. LWR Fuel Fabricated (metric tons)^(a)

Year	LWR-U	LWR-PU	Total
1975	1278.9	0.0	1278.9
1976	1471.8	0.0	1471.8
1977	1683.9	0.0	1683.9
1978	1864.2	0.0	1864.2
1979	2176.5	0.0	2176.5
1980	2648.6	144.1	2792.7
1981	3073.4	422.3	3495.7
1982	3442.9	618.8	4061.7
1983	4221.0	468.2	4689.2
1984	4709.4	530.4	5239.8
1985	5091.4	684.3	5775.7
1986	5678.5	733.4	6411.9
1987	6310.9	926.5	7237.4
1988	7005.1	1119.7	8124.8
1989	7657.5	1250.7	8908.2
1990	8488.6	1345.6	9834.2
1991	9255.4	1420.9	10676.3
1992	9883.4	1594.4	11477.8
1993	10700.7	1686.1	12386.8
1994	11513.7	1669.7	13183.4
1995	12282.8	1617.4	13900.2
1996	13205.6	1489.7	14695.3
1997	14083.0	1326.0	15409.0
1998	14963.8	1181.9	16145.7
1999	15569.1	1263.8	16832.9
2000	16411.4	1034.6	17446.0

a. Reference 7, based on assumptions summarized in Section 3.1.2.

TABLE 3.6. Shipments of Plutonium by Truck
to Fabrication Plants for LWR
Pu-Recycle Fuel^(a)

Year	MTHM per Year	Number of Shipments
1975	0.0	0.0
1976	0.0	0.0
1977	0.0	0.0
1978	0.0	0.0
1979	0.0	0.0
1980	6.11	12.2
1981	17.92	35.8
1982	26.25	52.5
1983	19.86	39.7
1984	22.50	45.0
1985	29.03	58.1
1986	31.11	62.2
1987	39.31	78.6
1988	47.50	95.0
1989	53.06	106.1
1990	57.08	114.2
1991	60.28	120.6
1992	67.64	135.3
1993	71.53	143.1
1994	70.83	141.7
1995	68.61	137.2
1996	63.19	126.4
1997	56.25	112.5
1998	50.14	100.3
1999	53.61	107.2
2000	43.89	87.8

a. Reference 7, based on assumptions summarized
in Section 3.1.2.

TABLE 3.7. Fresh Fuel Shipments by Truck to LWRs^(a)

Year	LWR-U		LWR-PU		Total	
	MTHM per Year	Number of Shipments	MTHM per Year	Number of Shipments	MTHM per Year	Number of Shipments
1975	1279	233	0	0	1279	233
1976	1472	268	0	0	1472	268
1977	1684	306	0	0	1684	306
1978	1864	339	0	0	1864	339
1979	2177	396	0	0	2177	396
1980	2649	482	144	26	2793	508
1981	3073	559	422	77	3496	636
1982	3443	626	619	113	4062	738
1983	4221	767	468	85	4689	853
1984	4709	856	530	96	5240	953
1985	5091	926	684	124	5776	1050
1986	5679	1032	733	133	6412	1166
1987	6311	1147	927	168	7237	1316
1988	7005	1274	1120	204	8125	1477
1989	7658	1392	1251	227	8908	1620
1990	8489	1543	1346	245	9834	1788
1991	9255	1683	1421	258	10676	1941
1992	9883	1797	1594	290	11478	2087
1993	10701	1946	1686	307	12387	2252
1994	11514	2093	1670	304	13183	2397
1995	12283	2233	1617	294	13900	2527
1996	13206	2401	1490	271	14695	2672
1997	14083	2561	1326	241	15409	2802
1998	14964	2721	1182	215	16146	2936
1999	15569	2831	1264	230	16833	3061
2000	16411	2984	1035	188	17446	3172

a. Reference 7, based on assumptions summarized in Section 3.1.2.

TABLE 3.8. Predicted Distribution of LWR Spent Fuel Reprocessing Among U.S. Facilities in the Near Term^(a)

Year	MTHM/yr ^(b)				
	AGNS	NFS	EXXON	New Capacity	Total
1977	500				500
1978	1000				1000
1979	1500				1500
1980	1500	300			1800
1981	1500	600			2100
1982	1500	600			2100
1983	1500	600	500		2600
1984	1500	600	1000		3100
1985	1500	600	1500		3600
1986	1500	600	1500	1000	4600
1987	1500	600	1500	1000	4600

a. See Table 3.9 for projected reprocessing schedules to year 2000.

b. Reference 7.

TABLE 3.9. Comparison of Reprocessing Requirements and Activities^(a)

Year	Actual LWR Reprocessing Load (MTHM)	LWR Fuel Reprocessed (MTHM)
1975	494.7	
1976	712.9	
1977	900.3	500.0
1978	1075.2	1000.0
1979	1259.5	1500.0
1980	1435.2	1800.0
1981	1631.5	2100.0
1982	1910.0	2100.0
1983	2298.4	2600.0
1984	2762.0	3100.0
1985	3284.0	3100.0
1986	3840.2	4100.0
1987	4397.3	5027.5
1988	4965.4	4965.4
1989	5577.8	5577.8
1990	6250.3	6250.3
1991	6961.0	6961.0
1992	7719.4	7719.4
1993	8515.0	8515.0
1994	9314.4	9314.4
1995	10153.8	10153.8
1996	10975.9	10975.9
1997	11744.9	11744.9
1998	12515.3	12515.3
1999	13267.7	13267.7
2000	14044.8	14044.8

a. Reference 7, based on assumptions summarized in Section 3.1.2.

TABLE 3.10. Shipments of Spent LWR Fuel
to Reprocessing Plants^(a)

Year	LWR-U		LWR-PU		Total	
	Truck	Rail	Truck	Rail	Truck	Rail
1977	409	103	0	0	409	103
1978	795	210	0	0	795	210
1979	1160	319	0	0	1160	319
1980	1350	389	0	0	1350	389
1981	1527	461	0	0	1527	461
1982	1381	437	97	31	1478	468
1983	1499	496	276	91	1775	587
1984	1652	575	390	136	2042	710
1985	1747	608	295	103	2042	710
1986	2287	834	323	118	2610	952
1987	2686	1029	402	154	3088	1183
1988	2619	1004	431	165	3050	1169
1989	2770	1120	523	212	3293	1332
1990	2943	1253	608	259	3551	1512
1991	3276	1395	679	289	3955	1684
1992	3506	1578	701	315	4207	1893
1993	3900	1755	740	333	4641	2088
1994	4070	1935	796	379	4866	2314
1995	4463	2122	842	400	5305	2522
1996	4685	2361	797	402	5482	2762
1997	5094	2567	772	389	5867	2956
1998	5307	2825	681	363	5988	3187
1999	5730	3058	605	323	6335	3381
2000	5870	3333	514	292	6384	3624

a. Reference 7, based on assumptions summarized in
Section 3.1.2.

TABLE 3.11. Projected Accumulation of Solidified LWR High-Level Wastes at Fuel Reprocessing Plants Before Shipment^(a)

Year	Annual Addition				Accumulation Through End of Year			
	Volume (Thousands of m ³)	Actinide Mass (MT)	Radio- activity (MCi)	Thermal Power (MW)	Volume (Thousands of m ³)	Actinide Mass (MT)	Radio- activity (MCi)	Thermal Power (MW)
1977	0.03	2.7	959	5	0.03	2.7	959	5
1978	0.06	5.5	2031	10	0.08	8.2	2516	12
1979	0.08	8.2	3202	15	0.17	16.4	4540	22
1980	0.10	9.9	4004	19	0.27	26.3	6506	31
1981	0.12	11.5	4817	23	0.39	37.8	8539	40
1982	0.12	12.1	4963	25	0.51	49.9	9994	48
1983	0.15	16.1	6240	33	0.66	66.0	12342	61
1984	0.18	19.7	7429	40	0.83	85.7	15058	77
1985	0.18	19.0	7369	39	1.01	104.7	16787	86
1986	0.23	24.8	9772	51	1.24	129.4	20550	105
1987	0.28	30.5	11981	63	1.53	160.0	25138	129
1988	0.28	30.4	11999	63	1.81	190.3	28100	145
1989	0.32	34.6	13580	72	2.12	224.9	31947	165
1990	0.35	39.1	15268	82	2.48	264.0	36342	189
1991	0.39	43.6	17029	92	2.87	307.5	41173	216
1992	0.44	48.1	18906	101	3.31	355.7	46434	244
1993	0.48	52.8	20886	111	3.79	408.5	52118	274
1994	0.53	57.9	22948	123	4.32	466.5	58202	306
1995	0.58	62.9	25083	134	4.89	529.3	64675	340
1996	0.62	67.4	27135	143	5.51	596.7	71374	374
1997	0.67	71.4	29062	152	6.18	668.0	78177	408
1998	0.71	75.0	30968	161	6.89	743.1	85118	441
1999	0.75	78.4	32810	168	7.64	821.5	92159	473
2000	0.80	82.1	34714	176	8.43	903.6	99401	505

a. Reference 7, based on assumptions summarized in Section 3.1.2.

TABLE 3.12. Containers of 10-yr-Old
LWR High-Level Waste
Shipped^(a)

<u>Year</u>	<u>Number of Containers</u>
1987	159
1988	318
1989	478
1990	573
1991	669
1992	669
1993	828
1994	987
1995	987
1996	1306
1997	1601
1998	1581
1999	1776
2000	1991
2001	2217
2002	2459
2003	2712
2004	2967
2005	3234
2006	3496
2007	3741
2008	3986
2009	4226
2010	4473

a. Reference 7, based on assumptions
summarized in Section 3.1.2.

TABLE 3.13. Projected Accumulation of Solidified LWR High-Level Wastes at a Federal Repository^(a)

Year	Annual Addition				Annual Number of Shipments	Accumulation Through End of Year			
	Volume (Thousands of m ³)	Actinide Mass ^(b) (MT)	Radio- activity (MCi)	Thermal Power (MW)		Volume (Thousands of m ³)	Actinide Mass ^(b) (MT)	Radio- activity (MCi)	Thermal Power (MW)
1987	0.03	2.7	115.29	0.4	13	0.03	2.7	115	0.4
1988	0.06	5.5	244.18	0.8	27	0.08	8.2	355	1.2
1989	0.08	8.2	385.05	1.3	40	0.17	16.4	728	2.5
1990	0.10	9.9	481.40	1.7	48	0.27	26.3	1184	4.1
1991	0.12	11.5	579.14	2.0	56	0.39	37.8	1724	5.9
1992	0.12	12.1	596.21	2.5	56	0.51	49.9	2264	8.2
1993	0.15	16.1	748.80	3.8	69	0.66	66.0	2940	11.6
1994	0.18	19.7	891.09	4.8	82	0.83	85.7	3739	16.0
1995	0.18	19.0	884.37	4.4	82	1.01	104.7	4507	19.8
1996	0.23	24.8	1173.11	5.5	109	1.24	129.4	5543	24.6
1997	0.28	30.5	1438.17	6.9	133	1.53	160.0	6812	30.7
1998	0.28	30.4	1440.14	7.0	132	1.81	190.3	8045	36.6
1999	0.32	34.6	1629.63	8.2	148	2.12	224.9	9432	43.6
2000	0.35	39.1	1831.86	9.5	166	2.48	264.0	10982	51.5
2001	0.39	43.6	2043.22	10.6	185	2.87	307.5	12699	60.3
2002	0.44	48.1	2268.51	11.6	205	3.31	355.7	14592	69.8
2003	0.48	52.8	2506.41	12.7	226	3.79	408.5	16669	80.1
2004	0.53	57.9	2753.65	14.0	247	4.32	466.4	18934	91.3
2005	0.58	62.9	3010.12	15.1	269	4.89	529.3	21391	103.4
2006	0.62	67.4	3256.91	16.0	291	5.51	596.7	24026	115.8
2007	0.67	71.4	3488.78	16.6	312	6.18	668.0	26819	128.5
2008	0.71	75.0	3718.36	17.1	332	6.89	743.1	29764	141.3
2009	0.75	78.4	3940.38	17.4	352	7.64	821.5	32850	153.9
2010	0.80	82.1	4169.76	17.8	373	8.43	903.6	36082	166.6

a. Reference 7, based on assumptions summarized in Section 3.1.2.

b. Note that uranium values are included in this group.

TABLE 3.14. Projected Accumulation of LWR Cladding Wastes at a Federal Repository^(a)

Year	Annual Addition				Accumulation Through End of Year			
	Volume ^(b) (Thousands of m ³)	Actinide Mass ^(c) (MT)	Radio- activity (MCi)	Thermal Power (MW)	Volume ^(b) (Thousands of m ³)	Actinide Mass ^(c) (MT)	Radio- activity (MCi)	Thermal Power (MW)
1978	0.10	0.24	2.9	0.032	0.10	0.24	3	0.03
1979	0.24	0.48	6.2	0.068	0.35	0.72	9	0.10
1980	0.35	0.72	9.7	0.107	0.70	1.45	17	0.19
1981	0.42	0.87	12.1	0.133	1.1	2.32	26	0.30
1982	0.49	1.01	14.6	0.161	1.6	3.33	37	0.42
1983	0.49	1.01	14.5	0.160	2.1	4.35	46	0.52
1984	0.60	1.26	17.3	0.192	2.7	5.60	57	0.64
1985	0.70	1.50	20.2	0.224	3.4	7.10	69	0.78
1986	0.70	1.50	20.6	0.227	4.1	8.60	80	0.91
1987	0.94	1.98	27.6	0.305	5.0	10.58	97	1.09
1988	1.2	2.43	33.8	0.373	6.2	13.01	117	1.32
1989	1.1	2.40	33.6	0.371	7.3	15.40	135	1.52
1990	1.3	2.69	37.7	0.417	8.6	18.10	155	1.74
1991	1.4	3.02	42.1	0.465	10	21.12	176	1.98
1992	1.6	3.36	47.0	0.519	12	24.48	200	2.24
1993	1.8	3.73	52.3	0.578	13	28.21	226	2.52
1994	2.0	4.11	58.0	0.641	15	32.32	254	2.84
1995	2.1	4.50	63.6	0.703	18	36.82	285	3.17
1996	2.3	4.90	69.8	0.771	20	41.72	318	3.53
1997	2.5	5.30	76.0	0.840	22	47.03	353	3.91
1998	2.7	5.67	82.1	0.906	25	52.70	390	4.31
1999	2.9	6.04	88.3	0.974	28	58.74	428	4.73
2000	3.0	6.41	94.5	1.042	31	65.15	468	5.16

a. Reference 7, based on assumptions summarized in Section 3.1.2.

b. Uncompacted volumes.

c. Note that uranium values are included in this group.

TABLE 3.15. Projected Accumulation of Fission Product
Tritium from LWR Fuel^(a)

Year	Annual Addition			Accumulation Through End of Year		
	Volume (m ³)	Radio- activity (MCi)	Thermal Power (Watts)	Volume (m ³)	Radio- activity (MCi)	Thermal Power (Watts)
1978	4.8	0.25	9	4.8	0.25	9
1979	10.1	0.53	19	14.9	0.76	27
1980	15.9	0.83	30	30.8	1.55	55
1981	19.9	1.04	37	50.7	2.50	89
1982	24.0	1.25	44	74.7	3.62	129
1983	24.5	1.28	46	99.2	4.70	167
1984	30.6	1.60	57	129.8	6.03	215
1985	36.3	1.89	67	166.1	7.60	270
1986	36.2	1.89	67	202.3	9.06	323
1987	48.1	2.51	89	250.3	11.07	394
1988	58.9	3.07	109	309.2	13.53	482
1989	58.9	3.07	109	368.1	15.86	565
1990	66.6	3.47	124	434.7	18.46	657
1991	74.8	3.90	139	509.5	21.35	760
1992	83.4	4.35	155	592.9	24.52	873
1993	92.6	4.83	172	685.5	28.00	997
1994	102.4	5.34	190	787.9	31.80	1132
1995	112.5	5.87	209	900.4	35.92	1279
1996	123.0	6.41	228	1023.4	40.36	1437
1997	133.2	6.95	247	1156.6	45.09	1605
1998	142.9	7.45	265	1299.5	50.06	1782
1999	152.5	7.95	283	1452.0	55.26	1967
2000	161.8	8.44	300	1613.8	60.66	2160

a. Reference 7, based on assumptions summarized in Section 3.1.2.

TABLE 3.16. Projected Accumulation of Fission Product Iodine from LWR Fuel^(a)

Year	Annual Addition				Accumulation Through End of Year			
	Volume (m ³)	Iodine Mass (kg)	Radio- activity (Curies)	Thermal Power (Watts)	Volume (m ³)	Iodine Mass (kg)	Radio- activity (Curies)	Thermal Power (Watts)
1976	0.0	0	0.0	0.0	0.0	0	0	0.0
1977	0.0	0	0.0	0.0	0.0	0	0	0.0
1978	0.04	104	14.2	0.009	0.04	104	14	0.009
1979	0.08	221	30.2	0.020	0.12	325	44	0.029
1980	0.13	348	47.6	0.031	0.25	673	92	0.061
1981	0.16	435	59.5	0.039	0.41	1108	151	0.100
1982	0.20	523	71.6	0.047	0.61	1631	223	0.147
1983	0.20	536	73.2	0.048	0.81	2167	296	0.195
1984	0.25	669	91.3	0.060	1.06	2836	388	0.255
1985	0.30	793	108.3	0.071	1.36	3629	496	0.326
1986	0.30	790	107.9	0.071	1.65	4419	604	0.397
1987	0.39	1050	143.4	0.094	2.05	5469	747	0.491
1988	0.48	1287	175.7	0.116	2.53	6756	923	0.607
1989	0.48	1287	175.8	0.116	3.01	8043	1098	0.722
1990	0.54	1455	198.6	0.131	3.55	9498	1297	0.853
1991	0.61	1634	223.0	0.147	4.16	11132	1520	1.000
1992	0.68	1822	248.7	0.164	4.84	12954	1769	1.163
1993	0.76	2024	276.3	0.182	5.60	14979	2045	1.345
1994	0.84	2238	305.4	0.201	6.44	17216	2350	1.546
1995	0.92	2458	335.5	0.221	7.36	19674	2686	1.767
1996	1.01	2688	366.9	0.241	8.36	22362	3053	2.008
1997	1.09	2911	397.4	0.261	9.45	25273	3450	2.269
1998	1.17	3122	426.2	0.280	10.62	28395	3876	2.550
1999	1.25	3332	455.0	0.299	11.87	31726	4331	2.849
2000	1.32	3535	482.9	0.318	13.19	35262	4814	3.167

a. Reference 7, based on assumptions summarized in Section 3.1.2.

TABLE 3.17. Projected Accumulation of Noble Gas Fission Products from LWR Fuel^(a)

Year	Annual Addition			Annual Number of Shipments	Accumulation Through End of Year		
	Number of Cylinders	Radio- activity (MCi)	Thermal Power (kW)		Number of Cylinders	Radio- activity (MCi)	Thermal Power (kW)
1978	25	3.9	6.4	4	25	4	6
1979	52	8.3	13.5	9	77	12	20
1980	82	13.1	21.4	14	159	24	40
1981	103	16.4	26.7	17	262	39	64
1982	124	19.7	32.1	21	386	57	92
1983	126	19.9	32.4	21	512	73	119
1984	155	24.3	39.5	26	667	93	151
1985	183	28.6	46.5	30	849	116	188
1986	183	28.8	46.8	31	1033	137	223
1987	244	38.4	62.5	41	1277	167	272
1988	299	47.0	76.5	50	1576	204	332
1989	299	46.9	76.4	50	1875	238	387
1990	337	52.8	85.9	56	2212	276	449
1991	378	59.1	96.2	63	2590	318	518
1992	422	66.0	107.3	70	3012	364	593
1993	469	73.4	119.4	78	3481	415	676
1994	519	81.2	132.2	86	3999	471	766
1995	569	89.2	145.1	95	4569	531	864
1996	623	97.6	158.9	104	5192	596	969
1997	676	106.1	172.6	113	5868	665	1082
1998	726	114.2	185.7	121	6594	738	1200
1999	777	122.4	199.1	129	7371	814	1325
2000	826	130.4	212.2	138	8197	894	1455

a. Reference 7, based on assumptions summarized in Section 3.1.2.

TABLE 3.18. Projected Accumulation of ^{14}C from Spent LWR Fuel^(a)

Year	Annual Addition (Ci)	Accumulation Through End of Year (Ci)
1978	266	266
1979	564	830
1980	889	1719
1981	1112	2830
1982	1337	4167
1983	1360	5526
1984	1679	7205
1985	1984	9188
1986	1987	11174
1987	2646	13818
1988	3240	17056
1989	3238	20293
1990	3653	23943
1991	4097	28038
1992	4570	32604
1993	5079	37679
1994	5619	43293
1995	6169	49458
1996	6751	56203
1997	7323	63518
1998	7864	71375
1999	8409	79776
2000	8941	88707

a. Reference 7, based on assumptions summarized in Section 3.1.2.

TABLE 3.19. Projected Accumulation of Beta-Gamma Wastes at Surface Burial Grounds^(a)

Year	Annual Addition			Accumulation Through End of Year		
	Uncompacted Volume (Millions of m ³)	Radio- activity (MCi)	Thermal Power (kW)	Uncompacted Volume (Millions of m ³)	Radio- activity (MCi)	Thermal Power (kW)
1975	0.014	0.04	0.49	0.035	0.09	1.1
1976	0.017	0.05	0.60	0.053	0.13	1.5
1977	0.021	0.06	0.70	0.073	0.17	2.0
1978	0.024	0.07	0.81	0.097	0.22	2.6
1979	0.028	0.08	0.92	0.125	0.27	3.2
1980	0.031	0.09	1.03	0.157	0.32	3.8
1981	0.051	0.10	1.19	0.208	0.38	4.5
1982	0.088	0.13	1.46	0.295	0.46	5.4
1983	0.118	0.15	1.78	0.413	0.55	6.5
1984	0.114	0.18	2.12	0.527	0.66	7.8
1985	0.132	0.21	2.49	0.659	0.79	9.2
1986	0.158	0.49	5.66	0.818	1.18	13.7
1987	0.175	0.55	6.37	0.993	1.57	18.3
1988	0.207	0.61	7.12	1.200	1.97	23.1
1989	0.242	0.69	8.02	1.442	2.40	28.1
1990	0.271	0.77	8.96	1.712	2.86	33.5
1991	0.297	1.28	14.93	2.009	3.77	44.1
1992	0.321	1.42	16.50	2.330	4.69	54.9
1993	0.356	1.56	18.11	2.686	5.64	66.0
1994	0.383	1.70	19.71	3.069	6.60	77.3
1995	0.399	1.83	21.32	3.468	7.58	88.7
1996	0.410	1.97	22.89	3.878	8.58	100.3
1997	0.413	2.10	24.41	4.291	9.58	111.9
1998	0.412	2.23	25.89	4.703	10.59	123.6
1999	0.412	2.35	27.38	5.115	11.62	135.2
2000	0.435	2.48	28.83	5.550	12.64	146.9

a. Reference 7, based on assumptions summarized in Section 3.1.2.

TABLE 3.20. Projected Accumulation of Plutonium Alpha (TRU, low level β - γ)
Wastes at a Federal Repository^(a)

Year	Annual Addition				Accumulation Through End of Year			
	Uncompacted Volume (Thousands of m ³)	Actinide Mass ^(b) (MT)	Radio- activity (MCi)	Thermal Power (kW)	Uncompacted Volume (Millions of m ³)	Actinide Mass ^(b) (MT)	Radio- activity (MCi)	Thermal Power (kW)
1975	0.3	0.003	0.0	0.0	0.001	0.01	0.1	0.1
1976	0.4	0.004	0.0	0.0	0.001	0.01	0.1	0.1
1977	0.6	0.005	0.1	0.1	0.002	0.02	0.2	0.2
1978	0.8	0.007	0.1	0.1	0.003	0.02	0.2	0.2
1979	1.0	0.009	0.1	0.1	0.004	0.03	0.3	0.3
1980	1.2	0.011	0.1	0.1	0.005	0.04	0.5	0.5
1981	6.6	0.058	0.7	0.6	0.011	0.10	1.1	1.0
1982	16.8	0.148	1.7	1.4	0.028	0.25	2.8	2.5
1983	24.4	0.215	2.5	2.1	0.053	0.46	5.2	4.7
1984	19.7	0.174	2.0	1.7	0.072	0.64	7.0	6.7
1985	22.7	0.201	2.3	2.0	0.095	0.84	9.0	8.9
1986	28.5	0.252	2.9	2.4	0.124	1.09	11.5	11.8
1987	31.0	0.273	3.2	2.7	0.155	1.36	14.2	15.0
1988	38.8	0.342	4.0	3.3	0.193	1.71	17.6	18.9
1989	46.4	0.410	4.8	4.0	0.240	2.12	21.6	23.7
1990	52.0	0.459	5.3	4.5	0.292	2.58	26.0	29.1
1991	56.4	0.498	5.8	4.8	0.348	3.07	30.7	35.1
1992	60.1	0.530	6.2	5.2	0.408	3.60	35.5	41.6
1993	67.3	0.594	6.9	5.8	0.476	4.20	40.9	48.9
1994	71.5	0.631	7.3	6.1	0.547	4.83	46.5	56.8
1995	72.0	0.636	7.4	6.2	0.619	5.47	51.9	65.0
1996	71.1	0.628	7.3	6.1	0.690	6.09	56.9	73.4
1997	67.4	0.595	6.9	5.8	0.758	6.69	61.4	81.6
1998	62.2	0.549	6.4	5.3	0.820	7.24	65.1	89.6
1999	57.6	0.508	5.9	4.9	0.877	7.75	68.2	97.3
2000	61.1	0.539	6.3	5.2	0.938	8.28	71.6	105.5

a. Reference 7, based on assumptions summarized in Section 3.1.2.

b. Note that uranium values are included in this group.

TABLE 3.21. Projected Accumulation of Alpha-Beta-Gamma (TRU, intermediate level β - γ) Wastes^(a)

Year	Annual Addition				Accumulation Through End of Year			
	Uncompacted Volume (Thousands of m ³)	Actinide Mass ^(b) (kg)	Radio- activity (MCi)	Thermal Power (kW)	Uncompacted Volume (Thousands of m ³)	Actinide Mass ^(b) (kg)	Radio- activity (MCi)	Thermal Power (kW)
1975	0.7	0.6	0.0	0.0	1.8	2	0.0	0.0
1976	1.0	0.9	0.0	0.0	2.8	2	0.0	0.0
1977	1.6	1.4	0.0	0.0	4.4	4	0.0	0.0
1978	2.1	1.8	0.1	0.6	6.4	6	0.13	0.6
1979	2.5	2.2	0.3	1.2	9.0	8	0.36	1.6
1980	3.0	2.7	0.4	2.0	12.0	11	0.69	3.0
1981	3.5	3.1	0.5	2.5	15.5	14	1.03	4.4
1982	4.0	3.5	0.7	3.0	19.5	17	1.40	5.9
1983	5.2	4.6	0.7	3.0	24.6	22	1.70	7.1
1984	7.2	6.3	0.8	3.7	31.8	28	2.11	8.7
1985	9.1	8.0	1.0	4.4	40.8	36	2.59	10.7
1986	9.7	8.6	1.0	4.4	50.6	45	2.98	12.1
1987	11.4	10.0	1.3	5.8	61.9	55	3.64	14.8
1988	13.4	11.9	1.6	7.2	75.4	67	4.45	18.0
1989	15.1	13.3	1.6	7.2	90.5	80	5.10	20.5
1990	17.4	15.4	1.8	8.1	107.9	95	5.85	23.4
1991	19.9	17.5	2.0	9.1	127.8	11	6.70	26.6
1992	22.2	19.6	2.2	10.1	150.0	132	7.64	30.3
1993	24.5	21.6	2.5	11.2	174.5	154	8.67	34.2
1994	26.8	23.7	2.8	12.4	201.3	178	9.80	38.6
1995	29.6	26.1	3.0	13.6	230.9	204	11.02	43.2
1996	32.1	28.3	3.3	14.9	263.0	232	12.33	48.2
1997	34.2	30.2	3.6	16.2	297.1	262	13.71	53.5
1998	36.0	31.8	3.9	17.4	333.1	294	15.15	59.0
1999	37.5	33.1	4.1	18.6	370.6	327	16.65	64.6
2000	38.8	34.2	4.4	19.8	409.4	361	18.21	70.4

a. Reference 7, based on assumptions summarized in Section 3.1.2.

b. Note that uranium values are included in this group.

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4.0 SELECTED GLOSSARY

4.0 SELECTED GLOSSARY

Back end of the fuel cycle: Includes reactors, spent fuel storage, fuel reprocessing, mixed-oxide fuel fabrication, and waste management.

Burial grounds: Areas designated for storage of containers of treated radioactive wastes by near-surface burial in geologic media.

Concepts: The basic methods by which waste management operations can be achieved. An example of a waste treatment concept is solidification of liquid high-level waste.

Decommissioning: Preparation of worn out or obsolete nuclear facilities for retirement. Decommissioning operations remove facilities such as reprocessing plants and burial grounds from service and reduce or stabilize radioactive contamination. Concepts include

- demolition and restoration to original conditions requiring no control
- partial demolition and fixation of residues
- minimal demolition followed by isolation and control of residues.

Deep continental geologic formations: Geologic media beneath the continents and isolated from biologic species and phenomena. Distinguished from ice sheets and sea floor geologic media.

Disposal: Operations designed to eliminate wastes from existence on earth or to permanently isolate them from mankind and his environs with no expectation of retrieval after emplacement. Isolation concepts include

- placement in subsurface geologic formations using technologies that offer no practical method for recovery
- emplacement into or beneath sea floors
- emplacement in ice sheets
- dispersion.

Elimination concepts include extraterrestrial disposal and transmutation.

Final storage: Storage operations for which a) no subsequent waste treatment or transportation operations are anticipated and b) conversion to disposal (i.e., termination of monitoring and human control) is considered possible.

The current concept for final storage is emplacement of wastes in geologic formations. The geologic formations may be on or near the earth's surface and augmented by engineering technology (e.g., burial grounds), or subsurface and augmented with engineering technology.

Head end of the fuel cycle: Mining, milling, enrichment, and fabrication of UO_2 fuel.

High-level waste: The fission product waste resulting from reprocessing spent fuel to separate uranium and plutonium from the fission products.

Immobilization: Treatment and/or emplacement of the wastes so as to impede their movement.

Interim storage: Storage operations for which a) monitoring and human control are provided and b) subsequent action involving treatment, transportation, or final disposition is expected.

Concepts for interim storage include bulk and unitized storage of solid, liquid, and gaseous wastes.

Alternative interim storage technologies include:

- tank storage of liquids
- canister storage in air-cooled vaults
- spent fuel storage in water basins.

Intermediate-level waste: Waste requiring some kind of action to protect personnel from radiation.

Isolation: A term encompassing both final storage and/or disposal in geologic formations.

Long-lived nuclides: Radioactive isotopes with half-lives greater than about 30 years. Most nuclides of interest to waste management have half-lives on the order of thousands to millions of years (^{239}Pu - 24,400 years; ^{99}Tc - 2.1×10^5 years; ^{129}I - 1.6×10^7 years).

Low-level waste: Wastes containing types and concentrations of radioactivity such that shielding to prevent personnel exposure is not required.

Non-high-level waste: Intermediate- or low-level waste.

Operations: Broad classification of waste management activities in terms of their basic function (e.g., waste storage, treatment, transportation or disposal).

Overpack: Secondary (or additional) external containment for packaged nuclear waste.

Primary wastes: As-generated forms and quantities of wastes.

Repository: A location containing wastes in storage or disposal.

Retrievability: Capability to remove waste from its place in final storage. The method and rate of removal and the subsequent location of the waste must satisfy retrievability criteria.

Secondary wastes: Forms and quantities of all wastes that result from applying waste treatment technologies to primary wastes.

Short-lived nuclides: Radioactive isotopes with half-lives no greater than about 30 years, e.g., ^{137}Cs and ^{90}Sr .

Solidification: Conversion of radioactive waste to a dry, stable solid.

Technologies: The specific methods for implementing concepts. An example is calcination of liquid high-level waste by using a spray calciner.

Transmutation: Conversion of a radioactive nucleus to another isotope by bombarding it with radiation or nuclear particles.

Transportation: Movement of materials between sites. Intra-site movement is not considered. Includes alternative methods for packaging, handling, and transport of waste materials and plutonium compounds. Concepts include all conventional methods of land and water transport required by the waste management system.

Transuranic elements: Elements with mass number greater than 92. They include neptunium, plutonium, americium, and curium.

Transuranic waste: Any waste material measured or assumed to contain more than a specified concentration (e.g., presently proposed as 10 nanocuries of alpha emitters per gram of waste) of transuranic elements.

Treatment: Operations intended to benefit safety or economy by changing the waste characteristics. Four basic treatment concepts are defined:

- volume reduction
- immobilization of radioactivity
- change of composition
- removal of radioactivity from the waste.