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Environmental Survey of the Reprocessing and Waste Management Portions of the LWR Fuel Cycle

A Task Force Report

Editors

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DISCLAIMER

FOREWORD

Pursuant to the National Environmental Policy Act of 1969 (NEPA), an environmental impact statement is prepared by the Nuclear Regulatory Commission in connection with issuance of a construction permit or an operating license for each light-water nuclear power reactor (LWR). These statements contain a detailed evaluation of the environmental impacts of construction and operation of the plant and a discussion of reasonable alternatives, as well as an overall assessment of the costs and benefits of the licensing action.

In November 1972, a document entitled, "Environmental Survey of the Nuclear Fuel Cycle" was published by the Directorate of Licensing of the Atomic Energy Commission (AEC). Its purpose was to establish a technical basis for informed consideration of environmental effects of the uranium fuel cycle in environmental impact statements for individual LWRs. In the Survey the nuclear fuel cycle was treated generically. To compensate for the consequent lack of specific site and design detail, estimates were made of effluent concentrations, radiation dose rates, and human population densities appropriate to the model fuel-cycle facilities. This approach was necessary because it was not possible to trace the fresh and spent fuel for an individual reactor through the fuel cycle and thus pinpoint environmental impacts at specific plants at specific points in time.

Comments on the Environmental Survey were solicited, and a hearing was held on February 1 and 2, 1973. The purpose of the hearing was to consider possible amendments to Appendix D of 10 CFR Part 50 which would, by rule, specify the environmental effects of the uranium fuel cycle that were to be factored into the assessment of costs and benefits in environmental impact statements for individual LWRs. Written comments were received in response to the Federal Register notice, and recommendations for improvement were offered during the hearings. After consideration of these comments, the AEC promulgated the final fuel-cycle rule (so-called Table S-3) on April 22, 1974 (39 FR 14188). The AEC indicated that the rule and survey would be re-examined from time to time to accommodate new information.* The same Table S-3 is now included in 10 CFR Part 51.

It was intended that, with the inclusion of environmental impacts from Table S-3, environmental impact statements for individual LWR's would represent a full and candid assessment of costs and benefits consistent with the legal requirements and spirit of NEPA. Consequently, the Survey was not intended as an analysis of alternatives, costs, and benefits of the entire uranium fuel cycle; i.e., it was not intended to be a complete environmental impact statement on the LWR fuel cycle.

*In this regard, the NRC staff is initiating a study designed to examine information that has developed since promulgation of the fuel-cycle rule in 1974 for the purpose of generally updating the rule.

On January 19, 1975, the AEC was abolished and its licensing and regulatory responsibilities transferred to the Nuclear Regulatory Commission. On July 21, 1976, the United States Court of Appeals for the District of Columbia Circuit decided Natural Resources Defense Council v. NRC and Aeschliman v. NRC, two cases involving judicial review of the fuel-cycle rule. The Court approved the overall approach and methodology of the rule. It found that, regarding most phases of the fuel cycle, the Survey did an adequate, even admirable, job of describing the processes involved. The Court noted that the Survey assembled data on consumption of resources, discussed the risks of accidents and other hazards in detail, and provided numerous references to the scholarly literature and technical reports in support of conclusions as to environmental impact. However, the Court found that the rule was inadequately supported by the record insofar as it treated two aspects of the fuel cycle--the impacts from reprocessing of spent fuel and impacts from radioactive waste management.

The Commission issued a General Statement of Policy (41 FR 34707, August 16, 1976) in response to the Court decisions. In that statement, the Commission announced its intention to reopen rulemaking proceedings on the environmental effects of the fuel cycle to supplement the existing record with regard to reprocessing and waste management, to determine whether the rule should be amended, and if so, in what respect. The Commission directed the staff to prepare on an expedited basis a well-documented supplement to the Survey to establish a basis for identifying environmental impacts associated with fuel reprocessing and waste management activities that are attributable to the licensing of a model LWR. This document is that Supplement.

In the original fuel-cycle rule the environmental impacts for fuel-cycle activities necessary for the support of a LWR were summarized in a Table S-3. These effects were also displayed by type of activity in Table S-3A of WASH-1248, April 1974.* In this Supplement, only the environmental impacts that would fall under the columns entitled "Reprocessing" (Column F), and "Waste Management" (Column G) of Table S-3A of WASH-1248 are addressed, consistent with the Court decisions and the Commission's General Statement of Policy.

"Waste management," as used in WASH-1248 and this Supplement, refers to the handling of wastes from post-fission operations in the fuel cycle, or other operations from which wastes arise and are shipped to some storage or burial facility. In WASH-1248 and this Supplement, wastes disposed of at the sites of their generation (e.g., tailings from mills) are included in the impacts from those operations (i.e., in other appropriate columns of Table S-3A). These wastes are mentioned in this Supplement in the description of the fuel cycle, but since these aspects of the original Survey were upheld by the Court, they are not addressed in detail here.

In the recently issued report "Final Generic Environmental Statement on the Use of Recycle Plutonium in Mixed Oxide Fuel in Light Water Cooled Reactors," NUREG-0002, (hereafter referred to as GESMO), five alternative fuel cycle options were evaluated, and three were reviewed in detail: no recycle, uranium-only recycle, and uranium

*"Environmental Survey of the Uranium Fuel Cycle," WASH-1248, USAEC, April 1974.

and plutonium recycle. The fuel-cycle rule (Table S-3), had as its base a modification of the uranium-only recycle process in which separated plutonium was stored for possible later use, rather than being recycled or treated as a waste stream as in GESMO. This Supplement considers the no-recycle fuel option in addition to the uranium-only recycle option. For the latter, plutonium is treated as a waste. The uranium-plutonium recycle option is treated in detail in GESMO and is beyond the scope of this Supplement.

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GLOSSARY

NOTE: The verbal conventions below, other than acronyms, apply solely to this Supplement, and are not intended to be definitions in a broader sense.)

AEC	Atomic Energy Commission; a former federal agency, disbanded by the Energy Reorganization Act of 1974..
AFR	Annual Fuel Requirement as used in WASH-1248.
AGNS	Allied-General Nuclear Services.
BNFP	Barnwell Nuclear Fuel Plant.
BNWL	Battelle Northwest Laboratories.
BWR	Boiling-Water Reactor.
CANDU	Canadian deuterium-uranium reactor.
Decommissioning	Taking out of service or ending active operations.
DF	Decontamination Factor.
Disposal	Permanent placement of wastes, without retrievability.
EPA	Environmental Protection Agency; a federal agency.
ERDA	Energy Research and Development Administration; a federal agency, successor to all nonregulatory functions of the Atomic Energy Commission.
Federal Repository	Federally operated disposal or storage facility for high-level and transuranic-contaminated wastes.
FRP	Fuel Reprocessing Plant.
GEIS	Generic environmental impact statement.
GESMO	Generic environmental impact statement for mixed oxide fuels (recycle plutonium in light-water-cooled reactors).
HEPA	High efficiency particulate air filters. Pleated fiberglass filters with high surface area and small pore size designed to remove aerosols with a minimum efficiency of 99.97% for 0.3 micrometer particles.
HLW	High-level wastes.
Impact	Used in this Supplement to mean only environmental impact.
Interim storage	Placement of wastes in an engineered temporary surround that will protect the public from radioactive release while a disposal location is developed.
LLW	Low-level wastes (containing minimal transuranic elements).
LWR	Light-Water Reactor.
MFRP	Midwest Fuel Recovery Plant.

MOX	Mixed oxide reactor fuel (containing both plutonium and uranium).
MT	Metric tons.
MTHM	Metric tons of heavy metal.
NEPA	National Environmental Policy Act of 1969.
NFS	Nuclear Fuel Services.
NRC	Nuclear Regulatory Commission; a federal agency, successor to regulatory functions of the Atomic Energy Commission.
Provisional storage	Storage in a planned disposal medium while disposal feasibility is assessed.
PWR	Pressurized water reactor.
RRY	Reference reactor year. A 1000-MWe reactor, assumed to be operating at 80% of its maximum capacity for one year. This is equivalent to AFR as used in the original WASH-1248, dated April 1974.
RSSF	Retrievable Surface Storage Facility.
SNM	Special Nuclear Material.
SRP	Savannah River Project.
SSCC	Sealed Storage Cask Concept.
Storage	Temporary placement of wastes, usually in a pool or other structure, that will allow cooling and radioactive decay in safety pending shipment to a reprocessing plant or repository.
SWU	Separative Work Unit.
TRU	Transuranic.
UNH	Uranium hexahydrate.

CHAPTER 1

APPROACH TO THIS SUPPLEMENT

This Supplement deals with the reprocessing and waste management portions of the nuclear fuel cycle for uranium-fueled reactors. The scope of the report is limited to the illumination of fuel reprocessing and waste management activities, and examination of the environmental impacts caused by these activities on a per-reactor basis. The approach is to select one realistic reprocessing and waste management system and to treat it in enough depth to illuminate the issues involved, the technology available, and the relationships of these to the nuclear fuel cycle in general and its environmental impacts.

1.1 Scope

The original Survey (WASH-1248) dealt with "the predominant uranium fuel cycle for uranium dioxide (UO_2) fueled, light-water moderated and cooled nuclear power plants (LWR)" (WASH-1248, p. S-1). While relying on some information from the literature based on assumptions of plutonium recycle, this Supplement deals explicitly with only those fuel cycles supportive of uranium-fueled LWRs.

Three fuel cycles are possible with LWRs (plus a number of variations), as discussed in detail in GESMO.*

- No recycling (with no reprocessing),
- Recycling of uranium only, and
- Recycling of both uranium and plutonium.

The environmental impacts from reprocessing and waste management are nearly identical for either of the recycle options. The wastes produced throughout the fuel cycle by the operation of uranium-fueled reactors are comparable for the two recycle options with the exception of the presence of plutonium as an additional waste in the uranium-only recycle option.

We have grouped the wastes from the nuclear fuel cycles into six major classes. Four of these arise from either recycle option: high-level wastes (HLW), transuranium-contaminated wastes (TRU), non-transuranium-contaminated wastes (often called low-level wastes--LLW), and contaminated facilities and large equipment. In addition, the uranium fuel cycles give rise to two special classes of wastes: spent fuel from the no-recycle option and unused plutonium from the uranium-recycle option.

*"Final Generic Environmental Statement on the Use of Recycle Plutonium in Mixed Oxide Fuel in Light Water Cooled Reactors: Health, Safety and Environment," NUREG-0002, USNRC, Washington, D.C., August 1976.

A number of tasks are required to take these wastes from the point and time of their production to their final sequestration. The release of radionuclides should be minimized to the extent practicable in all of these options. The operations are the subject of our major discussion of waste management and its environmental impacts.

Briefly, the operations through which each type of waste may need to pass include:

- Segregation from other in-process streams,
- Temporary storage to reduce radioactivity of short-lived isotopes,
- Treatment for shipment and/or disposal (solidification, stabilization, volume reduction, etc.),
- Packaging in appropriate shipping and/or disposal containers,
- Transportation to storage or disposal sites,
- Interim storage (if needed), and
- Disposal.

Within each of these are steps of handling, quality control, monitoring, etc., consistent with radiation safety both for workers and for the population at large. Wastes are produced at each fuel-cycle operation, and the wastes that are removed from the sites of production for storage and disposal are the subjects for discussion herein. The wastes of the greatest potential radiological environmental impact come either from the reactor itself (if spent fuel is the waste) or from the reprocessing plant (high-level and transuranium-contaminated wastes plus plutonium for the uranium-only recycle).

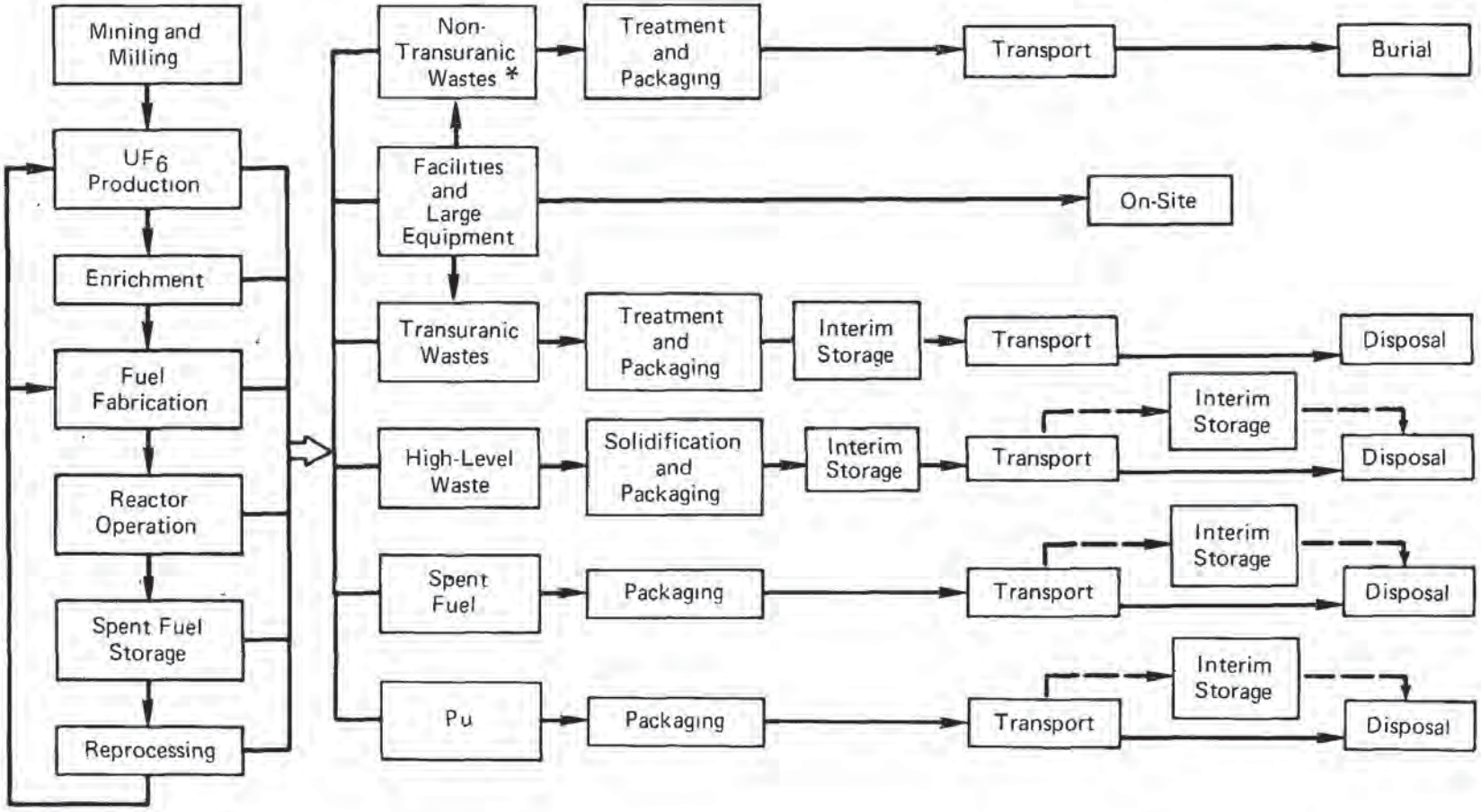
In a reprocessing plant, the spent-fuel elements are chopped into short segments, the fuel is dissolved in nitric acid, and the resultant solution is separated chemically to give (1) uranium in solution, (2) plutonium in solution, and (3) fission products and unwanted actinides in a waste stream. In this process, a number of other materials become contaminated and emerge as wastes of lesser radioactivity. All of these materials are made into a form suitable for shipment to the next step in the fuel cycle or to disposal. All operations required for these processes are parts of the reprocessing facility, and their impacts are included in the impacts described in this Supplement.

1.2 Approach

For the purposes of this Supplement we have selected a model or reference system for the fuel cycle, and in particular for the reprocessing and waste management portions of that system. This is shown schematically in Figure 1.1.

Each step in the reprocessing and waste management system has environmental impacts. These are frequently separate in both place and time and are discussed separately and sequentially.

Selection of the model system was governed by the following assumptions or contraints:



* Low-Level Wastes

FIGURE 1.1 - WASTE MANAGEMENT FLOW DIAGRAM

- The reprocessing and waste management system would receive material only from uranium-fueled LWR's.

- The industry supported by the selected system is mature; i.e., with sufficient facilities available and on-line for steady-state operation of all processing steps involved;

- The selected system is based on ERDA's currently contemplated waste management approach of geologic disposal as the final step;

- The technology must be available, (i.e., require no major scientific or engineering advances), and

- There must be sufficient information in the literature to permit illumination of the issues involved.

The system chosen for analysis here is a realistic and likely system. Because our dominant orientation is toward conservatism, this analysis is not always internally consistent (e.g., although the solidification system chosen is based on a glass product, a more dispersible calcine is used in many accident scenarios). All necessary elements of the system are discussed, however, and the analysis of each element is conservative to the best of our ability today.

Other systems with different environmental impacts can be postulated, but we believe such postulations would not alter our evaluation. Selection of the system for implementation can only be made through the kind of exhaustive analyses of alternatives that are planned by ERDA for its generic environmental statement and through the conclusions of ERDA and NRC programs for the years between now and 1985. These more complete analyses will reflect any synergistic or ameliorative effects that might result from assembling a different system from a different set of components than those chosen for the model used in this study, the sole objective of which is to describe likely environmental impacts.

On the basis of the best available information, the system selected for discussion here is representative of a system likely to be deployed, and the environmental impacts derived from discussion of that system are those to be expected from the fuel reprocessing and waste management portions of the fuel cycle supporting an individual reactor.

CHAPTER 2

SUMMARY OF IMPACTS OF REPROCESSING AND WASTE MANAGEMENT

In this Supplement we address environmental impacts listed in the columns entitled "Reprocessing" (Column F) and "Waste Management" (Column G) of Table S-3A. Impacts of normal and abnormal operations in reprocessing and waste management derived from the literature are normalized to show impacts resulting from one "reference reactor year" (RRY).

Reprocessing, as one of the operations in the nuclear fuel cycle, takes place on an individual site with a complex of interdependent facilities which are reviewed as a unit. A complete understanding of the management of nuclear wastes, however, requires a discussion of the operations through which they pass from the place and time of their generation to their disposal. Therefore, we separately discuss the operations at each facility in which the wastes are treated, stored, and prepared for shipment to offsite storage or disposal facilities. We further deal with the transportation of these wastes. The impacts of these operations are included in a separate discussion here and in the tables to follow (Tables 2.1 to 2.9) to aid in understanding of the management of radioactive wastes, but where they are already included in the appropriate fuel-cycle columns of Table S-3A they do not enter into the sum for waste management.

The impacts shown in the tables and discussed below are derived from the model fuel-cycle facilities used in GESMO,* and from environmental impact information derived from the literature. Some additional impact analyses were made for this Supplement; these were done where no previous analysis existed. While existing analyses are noted by reference to literature sources, these new analyses are given in Chapter 4 in their entirety.

Where no quantitative analysis was available, we have included a logical analysis, taking into account what is known of probabilities, consequences, and mitigating features. Where reasonable analysis leaves residual uncertainty we have so stated.

Finally, we have included as appendices a very brief discussion of ongoing federal programs in the area of nuclear waste management (Appendices B and C). These programs are directed at providing information, technology, and operating

*"Final Generic Environmental Statement on the Use of Recycle Plutonium in Mixed Oxide Fuel in Light Water Cooled Reactors: Health, Safety and Environment," NUREG-0002, USNRC, Washington, D.C., August 1976.

systems for the management of commercial nuclear wastes. They are based on considerable experience which has allowed the identification of questions, the assertion of hypotheses, and the pursuit of a direction dictated by the judgment that the hypotheses will prove correct. These programs should remove most of the remaining uncertainties within the next few years.

The key to our analysis is the selection of the technology for discussion. ERDA-76-43* provided the basis for that selection. In almost all instances, several technologies are presently available or very nearly available.

In general, the technologies selected are those presently being favored by either ERDA or the industry, and thus are the most likely to be eventually employed. The selection of these systems for this discussion does not reflect their endorsement by the NRC staff, since detailed licensing reviews have not been conducted.

Operating experience, either from ERDA military programs or from the industry, is available for portions of the reprocessing and waste management systems. Some of the waste management system components (e.g., high-level waste solidification) are extrapolations from laboratory or engineering-scale operations. Other parts of the system have never been used, and laboratory experiments related to these portions of the technology must be relied upon for operational parameters and performance predictions.

Overall, we find experience and experimental data adequate for the assessment of impacts of normal operations of all parts of the system. Abnormal occurrences have been analyzed for most parts of the system, but the bases for these analyses are varied, and not all pertinent accident sequences have been analyzed. For example, no quantitative evaluation or extensive systematic analysis, probabilistic or otherwise, of the consequences of accidents has been done for fuel-cycle activities as has been done for reactors (WASH-1400) (see Apps. B and C).** Sample events have been analyzed for all system components, and for one part (transportation) significant data on accidents for nonradioactive shipments were available and useful.

While intentional initiation of abnormal events--sabotage--has not been treated in any depth in the literature, it is possible to discuss the features of systems and procedures which have been designed to reduce the probability or consequences of sabotage. In general, these features do provide protection. Also, for some facilities, and the transportation of certain waste materials, additional protective measures are required by regulation. We therefore judge the risk from sabotage as adding only negligibly to the total risk.

Technologies for reprocessing and waste management are discussed in Chapter 4. Below is a brief summary of that discussion and its accompanying tabulation of environmental impacts. The structures of the two chapters are identical so that the reader may easily refer to details in the comparable section in Chapter 4.

*"Technical Alternatives for Managing Wastes from Reactors and Post-Fission Operations in the LWR Fuel Cycle," ERDA 76-43, USERDA, Washington, D.C., May 1976.
**Reactor Safety Study - An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants, WASH-1400 (NUREG 75/014), USNRC, Washington, D.C., October 1975.

2.1 Reprocessing

Spent fuel from uranium-fueled light water reactors (LWRs) contains considerable fissionable material--both uranium and plutonium--which may be recovered for use in new fuel. Reprocessing is an operation in the fuel cycle wherein spent fuel is processed to separate the usable uranium and plutonium from unwanted fission products (wastes) (see Sec. 4.1).

The Purex solvent extraction process has been the chosen method of fuel reprocessing for many years, and is the choice for evaluation here. The model was selected from the many plant variations possible and is the one used in GESMO.

In the fuel reprocessing plant (FRP) the spent fuel is chopped into pieces, the fuel is dissolved by nitric acid, and the resulting uranium, plutonium, and waste streams are converted to suitable physical and chemical forms either for disposal or for shipment and further use in the fuel cycle.

Variations on this technology have been in use by the AEC (now ERDA) and the industry for over 20 years. Environmental impacts derived from that experience but based on the model plant used in GESMO are listed in Tables 2.1 and 2.10 below, normalized to the contribution of a single RRY. Land commitments and resource use are typical of any industrial facility. Nonradiological effluents are small, and less than in comparable (e.g., chemical and petrochemical) industries. The primary radiological effluents are gaseous fission products and activation products from the spent fuel (krypton-85, iodine-131, carbon-14, etc.). These are the major effluents from the entire nuclear fuel cycle, and their assumed release gives a conservative estimate of total population dose commitment.* Total whole-body dose commitments to the world population from the reprocessing demanded by a single LWR in a year is about 600 person-rem for uranium-only recycle. The dose to the same population from natural background is about 300 million person-rem per year.

The literature regarding the environmental impacts of both normal and abnormal reprocessing operations is well developed. Some uncertainty remains regarding details of the design and operation of the plutonium conversion and HLW solidification facilities (see Sec. 4.2.2), but adequate experience at smaller scales is available to provide reasonable estimates of impacts at reprocessing plant sites.

2.2 High-Level Wastes

High-level wastes (HLW) produced at the reprocessing plant contain the majority of the highly radioactive fission products from the spent fuel. These wastes (see Ch. 3 and Sec. 4.2) require a system for their management which provides radiation shielding, protection against release, and a means of heat dissipation.

The conceptual system for HLW management includes the following steps: (1) short-term storage as liquid in tanks; (2) solidification; (3) short-term storage as a solid; (4) shipment to a repository; and (5) disposal. Provision for longer-term interim storage before disposal could be necessary and is included in our discussion.

*Pending EPA standards may further limit these releases.

Environmental impacts of normal operations in HLW management are well documented, and a number of accident scenarios have been analyzed which provide illustrative examples of impacts.

Impacts from HLW management are given for uranium-only recycle in the first column of Table 2(9). Land use is dominated by disposal and interim-storage facilities. Effluents from normal operations are small relative to the rest of the fuel cycle.

2.2.1 Tank Storage

Temporary storage of liquid HLW in tanks has been practiced for over 30 years. Tank designs have been improved (see App. E) and the most modern designs have proven virtually free of leaks and operational problems. Furthermore, the emerging philosophy of commercial reprocessing calls for solidification of HLW as soon as practicable after short periods of storage, eliminating the need for other than surge capacity in tank storage (in the event of process shutdown in the solidification facility).

The tank design discussed here is that of the BNFP,* comprising stainless-steel tanks in a stainless-steel-lined concrete vault with air-driven circulators and cooling coils for heat removal. Such tanks are an integral part of the reprocessing plant and all effluents from the tanks are treated with effluents from the rest of the plant. Their impacts are included among the impacts listed for reprocessing (Table 2.1).

2.2.2 Solidification

To prepare HLW for shipment and disposal, and to generally reduce the risk of its dispersal, it will be solidified as required by 10 CFR 50, App. F. A number of technologies exist for solidification, but reduction of the waste to a glass form has been favored by the AEC (and now ERDA) because of its advantages in reducing the risk of dispersal, and was selected as the model process for this Supplement.

The production of glass from liquid HLW is presently a two-step process (thus giving slightly more impact than the single-stage production of calcine), with a calcine produced first, which is then melted together with glass-forming materials to give the glass. The processes chosen for discussion herein are spray calcination (slightly more versatile than other calciners in that it can handle differing feed solutions) and in-can melting (the crucible is the disposal canister).

The product of the solidification process is a glass in a sealed canister ready for shipment, storage, or disposal. While selection of this process and product gives larger impacts for operation of the solidification facility, a less-dense solid such as calcine would give greater risk of dispersal in accidents during

*The Barnwell Nuclear Fuel Plant, located at Barnwell, S.C., and owned by Allied-General Nuclear Services (AGNS).

operations. Thus the waste forms giving the greater impacts are used here to provide conservatism in subsequent steps of the HLW management sequence.

The environmental impacts of operation of the solidification facility are displayed in Table 2.2 for purposes of illustration, but for the summation in Table 2.10 they are included in the overall impacts of the reprocessing facility.

Impacts shown for solidification, when compared with those for the entire reprocessing plant, indicate that moderate increments (up to 25-30% for heat and water) are added by that process to those from the main plant. Doses to the general population are a small fraction of (but are included in) the doses from reprocessing.

Analyses of representative accidents in the solidification facility show a small dose (a few tens of millirems) to an individual at the boundary of the facility attributable to those events (Sec. 4.2.2).

2.2.3 Interim Storage at the Reprocessing Plant

If the solidified HLW is not immediately shipped to a federal repository, storage capability at the reprocessing plant must be provided. Facilities similar to spent-fuel storage pools are chosen here for discussion. Shielding, confinement, and removal of decay heat are the major functions of the facility (Sec. 4.2.3).

During normal operations, only minor increments of heat release and water usage are added to the impacts of the reprocessing facility. Radioactivity from the outsides of canisters is separated from the water and handled with wastes from the reprocessing facility. Accidents at the storage facility have been analyzed, and significant impacts (i.e., releases) are not indicated.

2.2.4 Transportation

Following solidification or interim storage at the reprocessor, the solid wastes are to be shipped to a federal repository as required by 10 CFR 50, App. F. Impacts of that transportation are included in the discussion of transportation of all wastes (Secs. 2.9 and 4.9). These wastes will be transported in much the same manner as spent fuel is presently transported, and the impacts of HLW transportation are only a part of the impacts of transportation of all materials in the fuel cycle (see Tables 2.7 and 2.10).

2.2.5 Interim Storage at a Retrievable Surface Storage Facility

If final geologic disposal facilities are not available for receipt of solidified HLW ten years after their generation, a facility must be available for interim HLW storage. Such a facility is the retrievable surface storage facility (RSSF) (Sec. 4.2.5). The impacts for an RSSF are conservatively included in the summation.

Of three generic concepts available for such a facility, ERDA has chosen the Sealed Storage Cask Concept, which is also used in this Supplement. This concept is the least vulnerable to major accidents and requires the greatest commitment of land.

The design involves thick-walled, high-integrity overpacks inside concrete shields which provide shielding and channeling for natural-draft air cooling. After emplacement on a pad, the entire system is passive. Receiving and handling facilities are used for emplacement and monitoring. For purposes of environmental impact assessment, volumes were derived from an assumed glass form (maximum volume per RRY), and accidents were analyzed on the basis of an assumed calcine form (maximum dispersibility).

Land would be committed only temporarily, and effluents from normal operation would be small. Releases from cleanup and repair of canisters would give 5×10^{-3} person-rem per year to the U.S. population for a facility handling wastes from about 2000 RRY. Doses from the most serious postulated accident in the canister-handling facility were calculated to be about 85 millirem at 1000 meters from the stack.

2.2.6 Disposal at a Federal Repository

HLW disposal is assumed to be in bedded salt at a federal repository, the operation of which is described in Section 2.4 and given in greater detail in Section 4.4.

Impacts from routine operation of the facility before decommissioning (including sealing of the underground workings) are found to be small, and comparable to those of the RSSF. They are listed in Table 2.8 and are aggregated with other waste management impacts in Table 2.10, Column G.

2.3 Transuranic-Contaminated Wastes

Among the new nuclides produced in nuclear reactor fuel are a number which are heavier than uranium and are particularly toxic; some have very long half-lives (tens of thousands of years). Waste materials containing significant quantities of these long-lived elements will be confined and consigned to the federal repository.

Solid wastes contaminated with transuranics (TRU) are derived primarily from operation of the fuel reprocessing plant. Wastes included in this category are solidified liquids, filters, cladding hulls and other fuel hardware, and general trash.

Overall management will involve processing TRU waste to a stable form, packaging the product in a high-integrity container, storing the packages onsite at the FRP for up to 20 years, and finally shipping to a federal repository for long-term storage or geologic disposal.

Processing and interim storage facilities required for the management of TRU-contaminated wastes are substantially less complex and smaller than for the related major processing facility--the FRP.

For this discussion, the following steps are assumed to be required for the management of TRU wastes: 1) incineration of combustibles; 2) treatment (e.g.,

solidification) of noncombustibles; 3) packaging; 4) interim storage on site; 5) shipment to the repository; and 6) disposal in a manner similar to HLW. Impacts of these operations are given in Tables 2.3 and 2.8 and are included in Column F of Table 2.10. The environmental impacts of TRU waste management operations are a relatively minor (<10%) increment of those generated by the major process functions of the FRP.

2.3.1 Incineration of Combustible Wastes

A portion of the TRU wastes is combustible, presenting both a potential hazard and an opportunity for volume and hazard reduction by one of several oxidation processes. The process chosen for discussion herein is controlled-air incineration. This technology is under intensive investigation in the industry, and offers a number of advantages operationally (see Sec. 4.3.1).

Combustible waste will be placed in a conventional dual-chamber incinerator (modified for radioactive use). An off-gas cleaning system will contain any entrained or gaseous radioactive materials. Residual ash will be solidified in cement for storage and shipment to the repository.

Impacts for the operation are based on an assumed mixture of combustible wastes (paper, rags, plastics, and rubber). Account was taken of the fact that volatile fission products may be present in small quantities in TRU wastes from the fuel reprocessing plant.

Environmental impacts from operation of an incineration facility at the reprocessor are shown in Table 2.3, and are also included in the overall impacts of the reprocessor given in Table 2.10, Column F. Comparison shows that they are a small fraction of the overall impacts of the reprocessing plant.

Potential abnormal occurrences at the incinerator include fires and explosions. If an explosion breached the entire series of filters (despite attenuation by the volumes, fire screens, and filters), a release could occur. Assuming manual transfer to a backup system, a release of some 10^{-4} curie (alpha) might occur from an incinerator at a reprocessor--considerably smaller than normal releases from that plant.

2.3.2 Treatment of Noncombustible Wastes

Materials contaminated with transuranic elements will require treatment and packaging in preparation for shipment to the repository. These include (with the technology chosen for discussion herein):

- fuel element hulls and hardware (mixed with sand in welded steel containers),

- failed equipment (decontamination followed by size reduction and packaging in standard containers),

- ventilation filters (disassembly, compaction, and packaging),

- general trash (sorting and routing to other streams), and

- liquids and dispersible solids (incorporation in cement).

Some of these are conservative in that they tend to maximize onsite environmental impacts (e.g., size reduction for failed equipment). While these features would reduce the area required at the repository, no credit for such reduction was taken for analysis of impacts at the repository. Further, all the features are relatively consistent with industry and/or ERDA plans and requirements.

Environmental impacts of operations for treating noncombustible TRU wastes are given in Table 2.3. They are also included in impacts attributed to reprocessing in Table 2.10. They are based on analytical considerations and literature sources given in Section 4.3.2 of this Supplement. Under normal operating conditions these operations add very little to the impacts of the reprocessing plant.

Accidents in facilities for the treatment of noncombustible TRU waste will release only small amounts of radioactivity. One analysis (Sec. 4.3.2) indicates a release of less than 10^{-5} curie of plutonium. None of the treatment processes for noncombustible TRU wastes is a process of high energy content. Thus there is no driving force for dispersal of material in the event of accident. In addition, waste treatment and packaging is carried on within a confinement structure.

2.3.3 Interim Storage

Onsite storage of TRU may be required until a federal repository is available. A number of systems have been and are being used for the storage of such wastes. For discussion herein hulls are assumed to be stored in fuel-storage or HLW-storage basins, and other miscellaneous treated and packaged TRU is assumed to be stored retrievably in a vault equipped with a filtered ventilation system.

Both the waste containers and the storage facilities are assumed to be designed to high-integrity standards and should be virtually free of transferable radioactivity. Further, the wastes will have been fixed (though possibly dispersible) and packaged for shipment.

During normal operation of the facilities, environmental impacts are quite small (see Table 2.3) and are included in the summary column for reprocessing facilities in Table 2.10. Releases of radioactivity are expected to be less than 1 microcurie per RRY.

The abnormal occurrence analyzed for the storage of TRU wastes was spontaneous fire (e.g., hulls) in waste containers, assuming that the treatments described above had not taken place. From these postulated fires, a release of 1×10^{-4} to 3×10^{-4} Ci of plutonium would occur. If the TRU wastes had received the treatments described above, these releases would be reduced substantially.

2.3.4 Transportation

Mildly contaminated TRU wastes have been transported to burial grounds for many years (until recently). Hulls also have been buried. Transportation of these wastes to the repository is discussed in Sections 2.9 and 4.9, and the environmental impacts are shown to be small.

2.3.5 Disposal

TRU-contaminated waste materials are assumed to be disposed of at a federal repository. The requirements of that repository and the impacts of the disposal of TRU wastes are discussed in Sections 2.4 and 4.4. The impacts are included in Table 2.8 and in the summary column on waste management in Table 2.10.

2.4 Disposal of Long-Lived Wastes at a Federal Repository

HLW and TRU wastes comprise the materials from the nuclear fuel cycle which have sufficiently persistent biological hazards that they require special long-term considerations. (Plutonium and spent fuel as wastes are considered separately.) A number of conceptual alternatives for either eliminating these wastes or isolating them on earth are discussed in references cited in Section 4.4.

From those alternatives we have chosen for discussion deep emplacement in a stable geologic medium (bedded salt) under the continental U.S. This alternative was selected for two reasons: 1) it is the choice favored by many today; and 2) it has the greatest amount of substantive information available from which to summarize environmental impacts. While knowledge about other alternatives is more limited, we believe the salt-bed analysis to be reasonably representative of impacts that would result from any appropriately designed geological emplacement.

2.4.1 During Operation

For this discussion we use the technologies described in the literature for emplacement, namely conventional mining, at depth. The supporting operational facilities for this concept are very much like those at the RSSF, including handling, uncasking, inspection, cask repair, etc.

Sites selected will have to be from areas that have in the past been tectonically, hydrologically, and mechanically stable and free of mobile groundwater. In addition, the site must be chosen to minimize conflict with valuable natural resources. The judgment of geologists is that such sites exist, and a search for them is underway (see Apps. B and C).

The facility will be designed and the waste emplaced so as to keep the wastes and the geological formation below temperatures which could lead to nuclide migration or impair formation integrity. The mine will be constructed to prevent flooding and/or collapse during operation. Techniques exist for all these protections.

In Section 4.4 we discuss the salt medium, interaction of the wastes with that medium, possible mechanisms for nuclide migration, and some mechanisms that have been studied as possible initiators of untoward events. Engineering features will be built into the facility (e.g., ventilation for hydrogen and oxygen radiolysis products) to obviate problems from the processes that have been studied.

Operational (waste emplacement) lifetime of the facility, with about 2,000 acres of underground development, will be about 20 to 30 years. At that time the facility will be filled and sealed.

Environmental impacts of routine repository operation have been estimated and are displayed in Table 2.8 for a RRY. The commitment of land is about 2.0 acres per RRY, but after decommissioning of the repository, surface uses that do not involve disturbance of the deep formation can be allowed. Effluents (except large volumes of salt from excavation) are very small and not unlike those associated with a small industry. Radiological effluents come from routine package inspection and repair activities, and are quite small relative to those from other major fuel cycle facilities (e.g., reprocessing).

Accidents during operation have been analyzed and are much like those postulated for an RSSF. Because there is no driving force for dispersal, estimated releases are small; doses to an individual 1,000 meters from the stack are estimated as about 85 millirem from an individual event.

Fifty-year population dose commitments attributable to an RRY for the operation of a repository are of the order of 5×10^{-3} person-rem. Thermal outputs to the environment attributable to an individual RRY must be integrated over the lifetime of the waste (since that unit of waste continues to emit energy for that period of time independent of reactor operation), and these are given in Table 2.8. Because they are integral, they appear large, but the thermal flux at the surface is just a few times the normal geothermal flux at its maximum and gives a temperature rise of less than 0.1°F.

An impact that is unique to the repository (among fuel-cycle facilities) is alteration of the topography due to intermediate thermal expansion and to subsidence of the backfilled mine. A shallow surface depression of a few feet is anticipated ultimately.

2.4.2 Long-Term Risks

The length of time during which the hazards of transuranic nuclides must be protected against is the reason for the suggestion of a geologic repository, and is the center of most concerns expressed about management of nuclear wastes. The environmental impacts listed in Table 2.8 for geologic disposal in salt are those accrued during waste emplacement and up to the time the mine is sealed. Long-term impacts will be nonexistent if the repository performs as expected and maintains the wastes in isolation. The rationale for sequestration of the wastes in a geologic medium follows a simple line: since the geologic formation has been demonstrably

undisturbed for many millions of years, there is reason to believe that it will remain undisturbed into the future, even though mildly modified by placing the wastes into it. Further, a plastic medium such as salt will accommodate considerable geologic movement without exposing radioactive wastes to the readily available environment. Reasons for believing that the geological disposal system will perform for the full period needed for waste detoxification and will prevent significant releases are discussed in Section 4.4.

A number of types of events that could lead to the failure of geologic containment have been postulated: 1) rapid natural events; 2) natural geologic processes; 3) geologic processes resulting from waste emplacement; and 4) human intrusion, advertent or otherwise.

We cannot perform experiments to prove the capability of geological disposal in the usual scientific sense (i.e., a test of containment for a quarter of a million years), but even without the knowledge that experiment could bring, we can make some judgments based on the geologic record and on our knowledge of processes that would result from an initiating event. Probabilities of some initiating events have been estimated to be between 10^{-4} to 10^{-13} per year per model repository, which correspond to average occurrence rates ranging from once every 10,000 years to once every 10 trillion years. Not all such events lead to the same consequences. The more probable event of those studied (inadvertent drilling into the repository after loss of administrative control) leads to only minor releases and a means for repair is available (sealing the hole).

A less likely event (a major meteor strike on the repository, which has an average occurrence rate of once every 10 trillion years) could lead to a major release. While serious locally, the release of a fraction of the transuranics would be similar globally to releases of these materials that have already resulted from nuclear testing since WW II.

To arrive at some estimate of the consequences of waste release, two approaches have been taken. Radiotoxic hazard indices can be calculated (these are the amounts of water or air necessary to dilute a particular amount of radioactive material to maximum permissible concentrations), and modeling can be done of the transport of nuclides through the geosphere and biosphere, ingestion by man, and dose to man. The few discussions that exist regarding these approaches are summarized in Section 4.4. The radiotoxic hazard index analyses and the modeling studies that have been done indicate that consequences of all but the most improbable events will be small. Risks (probabilities times consequences) inherent in the long term for geological disposal will therefore also be small.

While our knowledge of the future is imperfect, the few analyses made to date provide some indication that no failure of geologic containment which has been analyzed will result in disaster to life on earth, or even to mankind. In fact, most events lead to very small effects. This merely confirms the logical argument that, in order for there to be a failure, the massive rock-salt formations themselves would require a tremendous removal of material which must either be done by very slow natural processes or by very high-energy events which are improbable. Salt is plastic, and under geologic pressures will flow to fill voids and reseal

intrusions, making improbable any channels through which radioactive material might spread to the environment.

Thus, from the information summarized above (and discussed in more detail in Sec. 4.4.2) we expect that possible releases in the long term (after repository decommissioning) will be negligible per reference reactor year.

2.5 Plutonium as a Waste

In the case of uranium-only recycle, the plutonium is not recovered and used for its fuel value. Instead it becomes a waste which must undergo much the same handling and disposal procedures as high-level or TRU wastes, with additional considerations for criticality control and long-term heat effects. Because plutonium has been viewed as a valuable energy source, there are few analyses of its disposal in the literature. In Section 4.5 of this Supplement a brief discussion is given of procedures that might be used for dealing with plutonium as a waste.

Two scenarios are discussed: 1) leaving the plutonium with the HLW (in solution and on through solidification); and 2) separating the plutonium but not purifying it beyond the first separation. In the first case (which presents considerable operational problems), the environmental impacts would be much the same as those for the handling of HLW except that the release of transuranic nuclides in the event of accident would be greater than for HLW. Environmental impacts of handling Pu in the HLW are included in Table 2.2 with HLW management. HLW that contains Pu will need to be disposed of in much the same manner as spent fuel, which is discussed in the section below.

If plutonium is separated from both uranium and the fission products, it can be handled in much the same way as presently proposed by the industry for handling recycled plutonium. Shipments of plutonium have been analyzed, but analyses in the literature do not extend to the disposal of packages of plutonium as waste. In Section 4.5, special considerations regarding plutonium disposal at the federal repository are presented. These include criticality, long-term heat effects, and an increase in possible long-term risks. Land-use commitments may increase by as much as 25% to accommodate long-term heating effects. Engineering features in the mine (neutron poisons, spacing, etc.) can provide measures that will mitigate criticality problems during or after the emplacement phase. The amount of plutonium in the repository for this fuel-cycle option would be a factor of 100 more than for disposal of HLW alone. Therefore the long-term risks associated with its disposal would be expected to be higher than those for disposal of HLW alone.

2.6 Spent Fuel as A Waste

For the model fuel cycle in which there is no recycle of uranium or plutonium, no reprocessing takes place, and the major waste would be spent fuel from reactors. In this case, the spent fuel can be managed with technology already in use for fuel storage and shipping. Only disposal at the federal repository will require features slightly different in kind from the technology discussed above. The disposal of

unprocessed spent fuel has received less attention and should therefore be considered in a less-developed technological state than the geologic storage of HLW.

Storage of spent fuel will require expanded fuel-pool storage; environmental impacts are given in Table 2.5.

Shipment of spent fuel in this case would be from the reactor to the repository and is similar to shipments from the reactor to the reprocessor. Its impacts are included in other tables (e.g., S-4 of WASH-1238).

Fuel for disposal could be packed in sealed steel containers with inert gas for heat exchange, which would be done to facilitate handling at the repository. Extra impacts due to this operation are small and are given in Section 4.6 and Table 2.5.

Disposal of spent fuel in salt in the federal repository poses the additional consideration (above HLW and TRU) of long-term heat load. This can be accommodated within the design of the repository, but there will be about 10 times as many canisters for disposal with spent fuel as would be the case with HLW. Incremental impacts of disposal of spent fuel (as opposed to HLW alone) are given in Table 2.8; these include increases in effluents and a minor decrease in land use. The major impact comes from the assumed release of gaseous fission products at the repository rather than at the reprocessor. This assumption is conservative since some of the gases will be contained in the mine, but no estimates of that containment have been made. In the long term, risks could be greater than those for HLW; the Pu content is higher by a factor of 100, and the fission products may be more leachable in the spent fuel than in solidified HLW after the container fails.

Without reprocessing, a number of types of wastes are eliminated from the nuclear fuel cycle, as are the impacts of HLW and TRU management. Changes in other impacts are beyond the scope of this document; we refer the reader to GESMO for a comparison of the fuel cycles, where the shifts in impacts throughout the fuel cycles are documented.

2.7 Nontransuranic Low-Level Wastes

A considerable volume of waste materials contaminated with fission products and/or activation products with relatively short half-lives is produced in fuel-cycle facilities. Maximum half-lives of the isotopes of concern are about 30 years, so that decay to innocuous levels occurs in hundreds of years, and isolation from the biosphere need not be for as great a period as for transuranic elements. Some of these wastes are of sufficient radioactivity to require shielding during handling and transportation, but most require no special shielding.

The variety of materials is similar to that in TRU wastes, and methods for handling, treating, and shipping are comparable. The wastes arise from almost all fuel-cycle facilities including reactors. Impacts of the treatment of these wastes at the reactor and during shipment to the burial ground are treated in each reactor environmental statement and are thus not included in WASH-1248 or in this Supplement.

However, impacts of disposal of wastes from the reactors themselves are included herein.

2.7.1 Treatment and Packaging

While the impacts of treatment and packaging facilities at the sites of LLW generation are included in the impacts of those sites, a brief discussion of LLW treatment appears in Section 4.7.1. Treatments include compaction, adsorption on clays and similar materials, solidification in cement or other materials, and packaging in containers that meet DOT specifications.

2.7.2 Transportation

Transportation of LLW to burial grounds is an operation with which there is considerable experience. Impacts of this activity are discussed in Sections 2.9 and 4.9, appear in Table 2.7, and are included in the overall transportation impacts in Column H of Table 2.10 and Table S-4 of WASH-1238.

2.7.3 Disposal

Because the lifetimes of radioisotopes in LLW are relatively short, the disposal technology selected for discussion herein is shallow land burial. This technology (described in Sec. 4.7.3) consists of excavating trenches tens of feet deep into soil, putting the as-shipped packages into the trenches, and covering with soil. The overburden provides both shielding protection and a buffer against erosion or contact with biota.

During site operation, administrative controls provide protection against inadvertent transmittal of radioactive materials outside the boundaries. Despite such controls, however, contaminated material was intentionally removed from a burial site at Beatty, Nevada, but the incident, which is the only one on record, led to no harm to the public. Nuclide content of LLW from the fuel cycle is generally low enough that similarly minimal effects could be expected from such acts in the future.

Protection against migration of waste nuclides after burial is provided mainly by selection of the site location to provide certain features. These include: 1) no permanent surface water; 2) good surface drainage with little potential for erosion; 3) the shallowest water table not less than 50 feet below the surface, with small fluctuations; and 4) soil of low permeability.

Environmental impacts of shallow land burial are derived from experience, tempered with expectations of improved performance from future sites. Impacts are tabulated in Table 2.6 and are included in the waste management column of Table 2.10. The major identified impact is the permanent commitment of about 0.1 acre of land per RRY.

2.8 Decontamination and Decommissioning of Nuclear Facilities

Major facilities and the equipment they contain become radioactive wastes at the end of their useful life. Further, if they are cleansed in part or entirely of radioactivity, the cleansing process produces additional wastes that require disposal. The procedure of taking a major facility out of service is termed decommissioning, and several options for that procedure are available under present regulations (see Sec. 4.8).

The procedure selected for discussion herein is that of decontaminating the facility of potentially hazardous radioactive material and removal of the radioactivity from the site. This procedure is conservative from a radiological perspective since greater impacts result from decontamination than might come from other options, but land-use commitments are reduced by the fact that the site is released for other uses.

Decontamination procedures vary according to the type of facility, as discussed in Section 4.8 and in more detail in the literature cited. In brief, they include such actions as: 1) equipment removal; 2) removal of cell liners; 3) removal of contaminated concrete; 4) chemical scrubbing; and 5) monitoring of all surfaces to assure adequate decontamination.

Environmental impacts of these activities are given in Table 4.30 for each facility, and again across the fuel cycle (normalized for the RRY) in Table 4.31. Wastes produced in these operations are included in TRU and LLW management discussions and calculations of impacts. Decontamination and decommissioning impacts were not included in facility-related impacts in the columns of Table S-3A, but are included in the "Waste Management" column of Table 2.10. Occupational exposure of workers during decontamination is a significant impact of these activities.

2.9 Transportation of Radioactive Wastes

Since there is no disposal of radioactive materials at fuel-cycle facilities, transportation of these materials to the point of their final disposition is an important activity. Transportation of radioactive materials to and from the power plant is the subject of WASH-1238 and is not covered here. All other shipments of radioactive wastes are included herein.

As discussed in Section 4.9 and detailed in references cited therein, most radioactive wastes are shipped in routine commerce and on conventional transportation equipment (truck, rail, barge), and are thus subject to ordinary transportation environments (accidents, road conditions, etc.). Primary reliance for safety is placed on packaging of radioactive materials so as to provide protection against the consequences of accidents as well as against radiation exposure of persons in the vicinity of packages being transported. In general, shielding is provided to reduce external radiation exposure rates, and packages are designed to remain intact in all except the most severe accident situations.

Environmental impacts of normal shipping operations include the diesel fuel expended, combustion gases emitted by transport vehicles, thermal energy given off by the radioactive cargo, and characteristic impacts of normal transportation. These are tiny increments in the overall transportation sector, but are noted in the discussion. Radiological exposures from normal fuel-cycle waste shipments are small (Table 4.35).

Accidents involving shipments of low-level TRU or non-TRU wastes can result in some release of radioactive material, but releases are of small impact and can be cleaned up at reasonable expense (see Sec. 4.9). While casks for spent fuel and/or HLW are massive and difficult to breach, accidents involving these casks have been analyzed. Cask design and the solid form of the waste protect against release of any of the solid radioactive material, but in the case of spent-fuel shipments some contaminated liquid coolant and volatile fission products can be released in very severe (low-probability) accidents. The analyses cited indicate that the risks are small.

2.10 Sabotage

Because the probabilities of an act of sabotage are difficult to estimate, prudence requires that measures be instituted to protect against sabotage attempts (deterrence) and against their likelihood of success (mitigation). Many of the physical features are an automatic consequence of shielding requirements, and others are specifically instituted to protect against accidents and/or sabotage.

Systems containing highly radiotoxic materials are designed to withstand severe accidents or natural phenomena. These same design features provide protection against sabotage. In addition, "vital" operations (those whose failure poses severe consequences) are performed within massive cells or buildings designed to maintain adequate confinement in the event of accident or violent natural phenomena.

Additional measures instituted to protect against diversion of materials or acts of sabotage include guard forces, physical and procedural access controls, detection aids, communication systems, and liaison with local law-enforcement agencies.

Regulations require that packages be designed to performance standards with respect to shielding, containment, heat transfer, and nuclear criticality. Packages containing the greatest sources of radioactivity are casks for spent fuel and HLW. For the uranium-only cycle, the waste plutonium (PuO_2 with 5% fission products) would also require shipment in shielded casks. Releases from these casks are difficult to effect because of their massiveness and leaktightness and the solid physical form of the contents.

Low-level and TRU wastes have large bulk and low radiological hazard, and thus provide little incentive for sabotage or potential for radiological release.

In Section 4.10.1 the nature of the threat from adversary groups is discussed qualitatively. While it is difficult to estimate probabilities of attack, the threat postulations and the historical record indicate that the likelihood of attack is not high.

Consequences of sucessful acts of sabotage have been analyzed for several scenarios and for several waste types (see Sec. 4.10.3). These include spent-fuel pools, HLW tanks, shipping packages, interim storage facilities, burial grounds, and repositories. While these analyses do not cover all possible scenarios, they indicate magnitudes of consequences that might arise from sabotage. The most serious consequences calculated are about one early death and up to 300 latent cancer deaths from the most serious and improbable transportation accidents.

While estimates of probabilities of future attempts at sabotage are necessarily speculative, the likelihood of successful sabotage of reprocessing and waste facilities and associated shipments is--with the application of protective measures as outlined above--judged to be sufficiently low to be negligible in the summation of environmental impacts (Table 2.10).

2.11 Tabulation and Summary of Environmental Impacts from Reprocessing and Waste Management

In Tables 2.1 to 2.10 are listed the environmental impacts of the reprocessing and waste management activities that support an individual reactor for one year of operation. The reference reactor and the fuel-cycle facilities used as models for this study are detailed in Chapter 3. The reprocessing and waste management operations are discussed in Chapter 4, and the environmental impacts tabulated here are derived from those discussions.

Table 2.10 is the aggregation of impacts for inclusion as Columns F and G and the "Total" column in Tables S-3A and S-3 of WASH-1248 and 10 CFR 51. For comparison, the original values are also given.

On the basis of the best available information, we conclude that (1) the reprocessing and waste management system discussed in Chapter 4 is representative of the system likely to be deployed; (2) the environmental impacts associated with that system, as summarized in Tables 2.1 through 2.10, are those to be expected from the reprocessing and waste management portions of the fuel cycle supporting an individual reactor; (3) these impacts (summarized in new Columns F and G of Table 2.10), are slightly different from those in WASH-1248, and (4) when these impacts are included in the total impacts of the uranium fuel cycle attributable to a single reactor (new Total column of Table 2.10), the total values are not substantially different from those in WASH-1248.

Because a different model reprocessing plant was used in this study (the GESMO model) than in WASH-1248, some variations are expected in the values in Table S-3A. These are seen in land and water use, in thermal and chemical effluents, and in radiological effluents.

In WASH-1248 no disposal of high-level waste was discussed, nor were plutonium or spent fuel as wastes considered. Furthermore, impacts at burial grounds for low-level wastes from reactor operations, reactor decommissioning, and other fuel cycle facilities were not included in the original Table S-3A. Thus an alteration in the "Waste Management" column of Table S-3A was anticipated and is indeed found in all the entries.

In arriving at the impacts tabulated below, a number of judgments had to be made on the basis of information which is not sufficiently complete for a detailed estimate of risk or impact. These judgments are noted in the text, and are only listed here.

- long-term risk of failure of the repository--negligible (based on some analyses which are indicative but not comprehensive)

- risks from sabotage--negligible (based on required design features and physical protection measures)

- impacts from accident sequences other than those analyzed--negligible (based on analyses of representative accidents)

- special risks from the disposal of spent fuel or separated plutonium--negligible (based on experience with these materials and the preliminary analyses in Chapter 4).

Total environmental impacts from the fuel cycle are shown in the last columns of Table 2.10 as derived from this Supplement and from the analysis in WASH-1248. These are discussed in some detail in WASH-1248 (pp. S-1 to S-30) and the discussion is not repeated here. However, for the convenience of the reader we reproduce Table S-3 from WASH-1248 (App. G) in its entirety, including the capsule notes of comparison in the final column of that table.

This Supplement introduces minor changes in many of the quantities in Table S-3, deriving from the different reprocessing plant model and from the fact that disposal operations in waste management are considered here but were not in WASH-1248. Only a few of the changes in this Supplement are substantial, however. These include:

Land use: An increase of about thirty acres per RRY in temporarily committed land results from the normalization procedures selected regarding that impact at the reprocessing plant (see footnote, Table S-1, WASH-1248). About 3 extra acres of land per RRY are permanently committed to disposal of the several types of wastes.

- Chemical effluents: Hydrogen chloride (HCl) gas is produced in the incineration of plastics, and is a new entry in the table, arising from our model of the waste management system. Amounts emitted are small, and no significant impacts are expected.

- Iodine release from the reprocessor:* The model of the process used gives larger iodine releases than those derived from the WASH-1248 model plant. The dose to the total U.S. population from iodine releases is about 1 person-rem per RRY (whole body), and about 260 person-rem per RRY to the thyroid.

- Carbon 14:* Derived from fuel reprocessing, this isotope was not identified in WASH-1248. This release causes a 111 person-rem dose commitment per RRY for the total U.S. population, which is still well below the total population dose of 2×10^7 person-rem per year from natural background.

- Buried solids: Radioactive wastes, upon disposal, although not released to the biosphere (i.e., the human environment) are placed in the geosphere and are thus included in the table. WASH-1248 did not include either the disposal of HLW and TRU contaminated wastes (these were merely stored in the original Survey) nor low-level wastes from reactors, which were buried. These wastes are included in Table 2.10, though we anticipate no environmental (radiological) impact from them.

2.12 Dose Summary

Table 2.11 presents a summary of the doses provided in the literature and given in each section of Chapter 4. Dose commitments are not presented in Table S-3A, but as they are a useful measure of radiological impact they are listed here to indicate the magnitudes of the impacts of the effluents summarized in that table. (For further comparison, the normal annual dose to an individual in the U.S. population from natural background ranges from about 80 mrem to about 150 mrem.) The table is provided as a convenient summary only. The cited text and references should be consulted for complete information.

Since numerous assumptions are made in the many references from which this table is compiled (e.g., population densities, atmospheric conditions, source terms, decontamination factors, integration time intervals, locations of maximum individuals, etc.), it is not proper to sum the doses to provide totals for the reprocessing and waste management portions of the fuel cycle.

2.13 Cumulative Impacts

The basis for the analysis used in this survey is the impact of a single model reactor operating for a single year. Reactors in fact operate for 30-year periods, and more than one reactor is operating at any given time. There is thus reason to inquire

*The Environmental Protection Agency is developing standards for limiting the dose to the public from the commercial nuclear power industry and its supporting uranium fuel-cycle activities. When effective, these federal standards would apply to impacts resulting from operations described in this Supplement.

about the cumulative impacts attributable to the reprocessing and waste management aspects of a number of reactors operating for their expected lifetimes. To provide a tabulation of such cumulative impacts, one must assume a model of a fuel cycle that includes (1) the nature of the operations in that cycle, (2) the growth of the industry, (3) the distribution of sites, (4) environmental parameters at the sites, and (5) demographic parameters surrounding the sites. Thus, a simple multiplication of the values in Table S-3 (or any other table herein) is not adequate to show cumulative impacts.

Such an analysis of cumulative impacts was performed in connection with the GESMO examination of fuel-cycle options. GESMO assumed an LWR industry of 507 reactors, and used an entire fuel-cycle model (not just the waste management and reprocessing aspects). Chapter VIII of that document calculates and tabulates at length the impact attributable to an industry of that size. The two fuel cycles treated in this Supplement are cases numbered five and six in GESMO; the tables developed from those cases (VIII A(5) and VIII A(6) of GESMO) are reproduced in Appendix G of this survey. Since this document is limited to waste management and reprocessing, we have performed no independent recalculation. However, the values given in those tables necessarily bound those of interest here. Two qualifications may be noted. First, the cumulative impact of a smaller number of reactors, say 200, cannot be derived by taking 200/507ths of the values stated in the tables, when other factors are taken into account, such as variations in reactor type, variations in time of initial operation of the reactors, the fact that reprocessing and waste management facilities will serve more than one reactor each, etc. Second, GESMO used slightly different assumptions about LWR characteristics than those used for this Supplement; an industry containing 507 reactors of the type assumed for this study--a single reactor which maximizes the enironmental consequences characteristic of BWRs and PWRs--would have environmental impacts slightly greater than those in the tables, which reflect a mixture of actual PWRs and BWRs.

There are presently 62 reactors licensed for operation and 143 construction permits authorized. Within the next five years, using present NRC projections, approximately 55 additional reactors will have completed review for operating licenses, and 63 more will have completed review for construction permits. At that time, if licenses and permits are granted, there will be 117 reactors operating and 151 under construction. These total substantially less than the 507 reactors assumed as a basis for the impacts given in the GESMO tabulation.

Table 2.1

SUMMARY OF ENVIRONMENTAL IMPACTS OF REPROCESSING PER RRY
(U-recycle only)

	Reprocessing
Natural Resource Use	
Land (acres)	
Temporarily committed[a]	32
Undisturbed area	28.5
Disturbed area	3.5
Permanently committed	0.12
Overburden Moved (millions of MT)	0.1
Water (millions of gallons)	
Discharged to air	6.6
Discharged to water bodies	54.8
Discharged to ground	---
Total	61.4
Fossil Fuel	
Electrical energy (thousands of MW-hr.)	4.0
Equivalent coal (thousands of MT)	1.5
Natural gas (millions of scf)[b]	28.6
Effluents - Chemical (MT)	
Gases (including entrainment)	
SO_x	5.4
NO_x	21.9
Hydrocarbons	0.5
CO	0.5
Particulates	0.6
Other Gases	
F^-	0.05
HCl	6E-4
Liquids	
$SO_4^=$	<0.02
NO_3^-	---
Fluoride	---
Ca^{++}	---
Cl^-	0.09
Na^+	<0.02
NH_3	---
Fe	---
Tailings Solutions (thousands of MT)	---
Solids	---
Effluents - Radiological (curies)	
Gases (including entrainment)	
Rn-222	---
Ra-226	---
Th-230	---
Uranium	3.9×10^{-5}

Table 2.1 (Cont'd)

	Reprocessing
Effluents - Radiological (curies) (Cont'd)	
C-14	24
Tritium (thousands)	18.1
Kr-85 (thousands)	400
I-129	0.03
I-131	0.83
Fission products	0.18
Transuranics	0.023
Liquids	None expected
Solids (buried)	
Other than high level	0.52
TRU and HLW	---
Effluents - Thermal (billions of Btu)	75.5

[a]The contributions to temporarily committed land are not prorated over 30 years, since the complete temporary impact accrues regardless of whether the plant services one reactor for one year or 57 reactors for 30 years.

[b]Thermal equivalent of fuel used at reprocessing plant site. Plant uses 200,000 gal #2 fuel oil and 2,100 gal propane per RRY. Combustion products of fuel oil and propane are included with process effluents in gaseous effluents.

Table 2.2

SUMMARY OF ENVIRONMENTAL IMPACTS OF MANAGEMENT OF HIGH-LEVEL WASTES
BEFORE DISPOSAL PER RRY
(U-only recycle)

	Solidification	Interim Storage at FRP	Interim Storage at RSSF
Natural Resource Use			
Land (acres)			
Temporarily committed	0.3		0.34
Undisturbed area	---		0.19
Disturbed area	0.3		0.15
Permanently committed	0.01	0.01	0.005
Overburden moved (millions of MT)	---	0.001	5E-4
Water (millions of gallons)			
Discharged to air	1.5	3.0**	0.28*
Discharged to water bodies	29.5	---	---
Discharged to ground	---	---	---
Total	31.0	3.0**	0.28*
Fossil Fuel			
Electrical energy (thousands of MW-hr.)	0.12	0.16**	0.09*
Equivalent coal (thousands of MT)	0.044	0.05**	0.03*
Natural gas (millions of scf)	---	---	---
Effluents - Chemical (MT)			
Gases (including entrainment)			
SO_x	---	---	0.02*
NO_x	0.63	---	0.02*
Hydrocarbons	---	---	0.02*
CO	---	---	---
Particulates	---	---	0.02*
Other Gases	---	---	---
Liquids	---	None expected	---
Solids	---	---	0.42*
Effluents - Radiological (curies)			
Gases (including entrainment)			
Rn-222	---	---	---
Ra-226	---	---	---
Th-230	---	---	---
Uranium	4.3E-11	---	---
C-14	---	---	---
Tritium (thousands)	0.79	---	---
Kr-85 (thousands)	---	---	---
I-129	5.9E-4	---	---
I-131	1.1E-5	---	---
Fission Products	2.6E-2	1E-5**	5E-5*
Transuranics	4.6E-5	---	---
Liquids	---	None expected	---
Solids (buried)			
Other than high level	---	0.52**	1.03*
TRU	---	---	---
HLW	---	---	---
Effluents - Thermal (billions of Btu)	9.2	63	29*

*Based on total for 20 years storage
**Based on total for 10 years storage

Table 2.3

SUMMARY OF ENVIRONMENTAL IMPACTS OF MANAGEMENT OF TRANSURANIC-CONTAMINATED WASTES BEFORE DISPOSAL PER RRY
(U-only recycle)

	Treatment		Interim Storage
	Combustible	Noncombustible	at FRP
Natural Resources Use			
Land (acres)			
Temporarily committed	0.02	---	0.1
Undisturbed area	---	---	---
Disturbed area	0.02	0.005*	0.1
Permanently committed	---	---	---
Overburden moved (millions of MT)	---	---	0.1
Water (millions of gallons)			
Discharged to air	4.0E-3	---	0.06
Discharged to water bodies	1.5E-2	---	---
Discharged to ground	---	---	---
Total	1.9E-2	---	0.06
Fossil Fuel			
Electrical energy (thousands of MWhr.)	0.016	0.2*	0.04
Equivalent coal (thousands of MT)	0.0058	0.07*	0.015
Natural gas (millions of scf)	0.002	---	---
Effluents - Chemical (MT)			
Gases (including entrainment)			
SO_x	0.0011	---	---
NO_x	0.0024	---	---
Hydrocarbons	---	---	---
CO	0.0011	---	0.0001
Particulates	1.6E-8	---	---
Other Gases			
F^-	---	---	---
HCl	6E-4	---	---
Liquids	---	None expected	---
Solids	---	None expected	---
Effluents - Radiological (curies)			
Gases (including entrainment)			
Rn-222	---	---	---
Ra-226	---	---	---
Th-230	---	---	---
Uranium	---	4E-10	---
C-14	---	---	---
Tritium (thousands)	---	---	---
Kr-85 (thousands)	---	---	---
I-129	7.2E-7	3E-7	---
I-131	1.7E-5	8E-6	---
Fission Products	0.0018	1.0E-6	---
Transuranics	1.1E-9	2E-7	1E-6
Liquids	---	None expected	---
Solids (buried)	---	None expected onsite	---
Effluents - Thermal (billions of Btu)	0.0019	---	0.5

*Prorated according to the fraction of the FRP devoted to noncombustible TRU waste treatment. Totals are included in the FRP total.

Table 2.4

SUMMARY OF ENVIRONMENTAL IMPACTS OF DECOMMISSIONING PER RRY
(Both cycles)

	U-Only Recycle		No Recycle
Natural Resource Use			
Land (acres)	---	Not applicable[a]	---
Water (millions of gallons)			
Discharged to air	0.0034		0.004
Discharged to water bodies	0.039		0.049
Discharged to ground	0.16		0.16
Total	0.20		0.22
Fossil Fuel			
Electrical energy (thousands of MW-hr.)	b		b
Equivalent coal (thousands of MT)			
Natural gas (millions of scf)			
Effluents - Chemical (MT)			
Gases (including entrainment)	---	None expected	---
Liquids	---	None expected	---
Solids	---	None expected	---
Effluents - Radiological (curies)			
Gases (including entrainment)			
Rn-222	---		---
Ra-226	4.5E-7		5.3E-7
Th-230	4.5E-7		5.3E-7
Uranium	7.3E-6		7.9E-6
C-14	---		---
Tritium (thousands)	---		---
Kr-85 (thousands)	---		---
I-129	---		---
I-131	---		---
Fission Products	2.9E-3		---
Transuranics	2.9E-4		---
Liquids			
Uranium & daughters	5.4E-6		5.4E-6
Fission and Activation Products	4.5E-6		5.9E-6
Ra-226	---		---
Th-230	---		---
Th-234	---		---
Ru-106	---		---
Tritium (thousands)	---		---
Solids (buried)	---	None onsite	---
Effluents - Thermal (billions of Btu)	---		---

[a] Actually has negative impact (land is released for other use) but this was not estimated.

[b] Expected to be small on a normalized basis.

Table 2.5

SUMMARY OF ENVIRONMENTAL IMPACTS OF MANAGEMENT OF SPENT FUEL AS A WASTE BEFORE DISPOSAL PER RRY

Natural Resources Use	Storage		Packaging
Land (acres)			
Temporarily committed	---		---
Undisturbed area	---		---
Disturbed area	---		---
Permanently committed	0.1		0.1
Overburden moved (millions of MT)	---		---
Water (millions of gallons)			
Discharged to air	11		---
Discharged to water bodies	---		---
Discharged to ground	---		
Total	11		---
Fossil Fuel	---	None expected	---
Effluents - Chemical (MT)	---	None expected	---
Effluents - Radiological (curies)			
Gases (including entrainment)	---		---
Rn-222	---		---
Ra-226	---		---
Th-230	---		---
Uranium	---		---
C-14	---		---
Tritium (thousands)	---		---
Kr-85 (thousands)	0.35		0.35
I-129	---		---
I-131	---		---
Fission Products	---		---
Transuranics	---		---
Liquids	---	None expected	---
Solids (buried)			
Other than high level	70[a]		70[a]
TRU and HLW	---		---
Effluents - Thermal (billions of Btu)	89		1.0

[a]At burial grounds.

Table 2.6

SUMMARY OF ENVIRONMENTAL IMPACTS OF DISPOSAL OF LOW-LEVEL WASTES
(Non-TRU) PER RRY

Natural Resource Use	Burial
Land (acres)	
Temporarily committed	0.3
Undisturbed area	0.2
Disturbed area	0.1
Permanently committed	0.1
Overburden Moved (millions of MT)	0.001
Water (millions of gallons)	0.0[b]
Fossil Fuel	0.0[a]
Effluents - Chemical (MT)	
Gases (including entrainment)	0.0[b]
Liquids	0.0[b]
Solids	0.0[b]
Effluents - Radiological (curies)	
Gases (including entrainment)	0.0[c]
Liquids	0.0[c]
Solids (buried)	
Other than high level	4700[d]
TRU and HLW	0.0
Effluents -Thermal (billions of Btu)	0.0[b]

(a) Contribution negligibly small compared with other waste handling steps.
(b) No process effluents. Natural runoff of precipitation may occur, but will not result in radiological transport.
(c) No releases of contaminated liquids are expected at model site.
(d) Value does not include contribution from decommissioning which has not been estimated.

Table 2.7

SUMMARY OF ENVIRONMENTAL IMPACTS OF TRANSPORTATION OF RADIOACTIVE WASTES PER RRY
(U-only recycle)

Natural Resource Use	Transportation
Land (acres)	None
Water (millions of gallons)	None
Fossil Fuel	
Electrical energy (thousands of MW-hr.)	None
Equivalent coal (thousands of MT)	0.016
Natural gas (millions of scf)	
Effluents - Chemical (MT)	
Gases (including entrainment)	
SO_x	0.045
NO_x	0.62
Hydrocarbons	0.062
CO	0.38
Particulates	0.022
Other Gases	---
Liquids	None
Solids	None
Effluents - Radiological (curies)	None
Effluents - Thermal (billions of Btu)	0.014

Table 2.8

SUMMARY OF ENVIRONMENTAL IMPACTS OF DISPOSAL OF LONG-LIVED WASTES
AT A FEDERAL REPOSITORY PER RRY
(Both cycles)

Natural Resource Use	U-Only Recycle (HLW + TRU + Pu)	No Recycle
Land (acres)		
Temporarily committed	2.5	2.0
Undisturbed area	2.5	2.0
Disturbed area	0.031	0.025
Permanently committed	2.5	2.0
Overburden Moved (millions of MT)	-	-
Water (millions of gallons)		
Discharged to air	0.1	0.1
Discharged to water bodies	-	-
Discharged to ground	0.8	0.8
Total	0.9	0.9
Fossil Fuel		
Electrical energy (thousands of MW-hr.)	0.53	0.53
Equivalent coal (thousands of MT)	0.19	0.19
Natural gas (millions of scf)	3.3	3.3
Effluents - Chemical (MT)		
Gases (including entrainment)		
SO_x	0.0096	0.0096
NO_x	0.011	0.011
Hydrocarbons	1.1E-4	1.1E-4
CO	0.007	0.007
Particulates	2.4E-5	2.4E-5
Other Gases		
F^-	-	-
HCl	0.013	0.013
Liquids	--- None expected	---
Solids	--- None expected	---
Effluents - Radiological (curies)		
Gases (including entrainment)		
Rn-222	0.0071	0.0071
Ra-226	-	-
Th-230	-	-
Uranium	-	-
C-14	-	19
Tritium (thousands)	6.8E-9	14
Kr-85 (thousands)	1.1E-7	290
I-129	-	1.3
I-131	-	-
Fission Products	4.2E-5	4.2E-5
Transuranics	1.1E-3	1.1E-3
Liquids	--- None expected	---
Solids (buried)		
Other than high level	---	---
TRU and HLW	1.1E+7	1.1E+7
Effluents - Thermal (billions of Btu)	59[a]	59[a]

[a]Thermal output from waste from one RRY integrated over time of significant heat release.

Table 2.9

SUMMARY OF IMPACTS OF MANAGEMENT OF WASTES DURING ALL OPERATIONS PER RRY (Both Cycles)[a]

Natural Resource Use	Total Waste Management Impact (U-Only Recycle)	Total Waste Management Impact (No Recycle)
Land (acres)		
Temporarily committed	3.6	2.3
Undisturbed area	2.9	2.2
Disturbed area	0.71	0.13
Permanently committed	2.6	2.3
Overburden Moved (millions of MT)	0.10	0.001
Water (millions of gallons)		
Discharged to air	4.9	11
Discharged to water bodies	30	-
Discharged to ground	0.8	0.8
Total	36	12
Fossil Fuel		
Electrical energy (thousands of MW-hr.)	1.2	0.53
Equivalent coal (thousands of MT)	0.40	0.19
Natural gas (millions of scf)	3.3	3.3
Effluents - Chemical (MT)		
Gases (including entrainment)		
SO_x	0.031	0.0096
NO_x	0.66	0.011
Hydrocarbons	0.02	1.1E-4
CO	0.0082	0.007
Particulates	0.02	2.4E-5
Other Gases		
F^-	-	-
HCL	0.014	0.013
Liquids	--- None expected	---
Solids	0.42	-
Effluents - Radiological (curies)		
Gases (including entrainment)		
Rn-222	0.0071	0.0071
Ra-226	-	-
Th-230	-	-
Uranium	4.4E-10	-
C-14	-	19
Tritium (thousands)	0.79	14
Kr-85 (thousands)	1.1E-7	290
I-129	5.9E-4	1.3
I-131	3.6E-5	-
Fission Products	0.028	4.2E-5
Transuranics	0.0011	1.1E-3
Liquids	--- None expected	---
Solids (buried on site)		
Other than high level (shallow)	4700	4700
TRU and HLW (deep)	1.1E+7	1.1E+7
Effluents - Thermal (billions of Btu)	161	149

[a]Includes impacts of all operations (except transportation) in the management of all kinds of wastes from the time and point of origin (after separation at the reprocessing plant) to the time and point of disposal. This is not the same as Column G in Table 2.10, which includes impacts that occur at sites other than the site of origin. Impacts of waste-management operations at the site of origin are included in the appropriate columns of Table S-3A of WASH-1248 and are included in the "Total" column in Table 2.10.

Table 2.10

SUMMARY OF IMPACTS OF REPROCESSING AND WASTE MANAGEMENT PER RRY
(Columns F, G, and H of Table S-3A)

Natural Resource Use	F[d] Reprocessing (NUREG-0116)	F[d] Reprocessing (WASH-1248)	G[a] Waste Management (NUREG-0116)	G Waste Management (WASH-1248)	H[c] Transportation (NUREG-0116)	H Transportation (WASH-1248)	Fuel Cycle Totals[a,b] (NUREG-0116)	Fuel Cycle Totals[a,b] (WASH-1248)
Land (Acres)								
Temporarily Committed	32[h]	3.9	3.2	-		-	94	63
Undisturbed Area	28.5	3.7	2.9	-	-	-	73	45
Disturbed Area	3.5	0.2	0.28	-	-	-	22	18
Permanently Committed	0.12	0.03	2.6[e]	0.2	-	-	7.1	4.6
Overburden moved (millions of MT)	0.1	-	0.0015	-	-	-	2.8	2.7
Water (millions of gal.)								
Discharged to air	6.6	4.0	0.38	0.13	-	-	159	156
Discharged to water bodies	54.8	6.0	0.051	0.13	-	-	11,090	11,040
Discharged to ground	-	-	0.96	-	-	-	124	123
Total Water	61.4	10.0	1.4	0.26	-	-	11,373	11,319
Fossil Fuel								
Electrical energy (thousand MW-hr.)	4.0	0.45	0.62	.0077	-	-	321	317
Equivalent coal (thousand MT)	1.5	0.16	0.22	.003	0.016	-	117	115
Natural Gas (million scf)	28.6	-	3.3	-	-	-	124	92

[a] Maximized for either of the two cycles (U-only and no recycle).

[b] Including columns A-E of Table S-3A of WASH-1248.

[c] For wastes only.

[d] Differences between NUREG-0116 and WASH-1248 estimates of reprocessing impacts are attributable to the use of a new model plant for this Supplement.

[e] Disposal included here did not appear in WASH-1248.

[f] Not released to the environment.

[g] Major radionuclide releases in the Waste Management column are attributable to the disposal of spent fuel (no-recycle option) and the conservative assumption of complete release of gaseous nuclides in the geologic repository.

[h] The contributions to temporarily committed land are not prorated over 30 years, since the complete temporary impact accrues regardless of whether the plant services one reactor for one year or 57 reactors for 30 years.

Table 2.10 (Continued)

Natural Resource Use	F[d] Reprocessing (NUREG-0116)	F[d] Reprocessing (WASH-1248)	G[a] Waste Management (NUREG-0116)	G Waste Management (WASH-1248)	H[c] Transportation (NUREG-0116)	H Transportation (WASH-1248)	Fuel Cycle Totals[b] (NUREG-0116)	Fuel Cycle Totals[b] (WASH-1248)
Effluents								
Chemical (MT)								
Gases (MT)								
SO_x	5.4	6.2	0.030	-	0.045	-	4,400	4,400
NO_x	21.9	7.1	0.031	-	0.62	2.6	1,190	1,177
Hydrocarbons	0.5	0.02	0.02	-	0.062	-	14	13.5
CO	0.5	0.04	0.007	-	0.38	-	29.6	28.7
Particulates	0.6	1.6	0.02	-	0.022	-	1,154	1,156
Other Gases								
F^-	0.05	0.11	-	-	-	-	0.67	0.72
HCl	6E-4	-	0.013	-	-	-	0.14	-
Liquids								
$SO_4^=$	< 0.02	0.4	-	-	-	-	9.9	10.3
NO_3^-	-	0.9	-	-	-	-	25.8	26.7
Fluoride	-	-	-	-	-	-	12.9	12.9
Ca^{++}	-	-	-	-	-	-	5.4	5.4
Cl^-	0.09	0.2	-	-	-	-	8.5	8.6
Na^+	< 0.02	5.3	-	-	-	-	12.1	16.9
NH_3	-	-	-	-	-	-	10.0	11.5
Tailings Solutions (thousands)	-	-	-	-	-	-	240	240
Fe	-	-	-	-	-	-	0.4	0.4
Solids	-	-	0.42	-	-	-	91,000	91,000

Table 2.10 (Continued)

Natural Resource Use	F[d] Reprocessing (NUREG-0116)	F[d] Reprocessing (WASH-1248)	G[a] Waste Management (NUREG-0116)	G Waste Management (WASH-1248)	H[c] Transportation (NUREG-0116)	H Transportation (WASH-1248)	Fuel Cycle Totals[b] (NUREG-0116)	Fuel Cycle Totals[b] (WASH-1248)
Effluents (Cont'd.)								
Radiological (curies)								
Gases (including entrainment)								
Rn-222	-	-	0.0071	-	-	-	74.5	74.5
Ra-226	-	-	5.3E-7	-	-	-	0.02	0.02
Th-230	-	-	5.3E-7	-	-	-	0.02	0.02
Uranium	0.000039	-	7.9E-6	-	-	-	0.034	0.032
Tritium (thousands)	18.1	16.7	149	-	-	-	18.1	16.7
Kr-85 (thousands).	400	350	290[g]	-	-	-	400	350
I-129	0.03	0.0024	1.3[g]	-	-	-	1.3	0.0024
I-131	0.83	0.024	-	-	-	-	0.83	0.024
Fission Products	0.18	1.0	0.003	-	-	-	0.021	1.0
Transuranics	0.023	0.004	0.0014	-	-	-	0.024	0.004
C-14	24	-	19[g]	-	-	-	24	-
Liquids								
Uranium & Daughters	-	-	5.4E-6	-	-	-	2.1	2.1
Fission & Activation Products	-	-	5.9E-6	-	-	-	5.9E-6	-
Ra-226	-	-	-	-	-	-	0.0034	0.0034
Th-230	-	-	-	-	-	-	0.0015	0.0015
Th-234	-	-	-	-	-	-	0.01	0.01
Tritium (thousands)	-	2.5	-	-	-	-	-	2.5
Ru-106	-	0.15	-	-	-	-	-	0.15
Solids (buried onsite)[f]								
Other than high level (shallow)	0.52	-	4,700	-	-	-	5,300	601
TRU & HLW (deep)	-	-	1.1E+7	-	-	-	1.1E+7	-
Thermal (billions of Btu)	75.5	61	88	1.0	0.014	0.03	3,462	3,360

Table 2.11

DOSE COMMITMENTS - SUMMARY

(Normal Operations)

	Dose	Basis	NUREG-0116 Section	Reference
Reprocessing	330 person-rem/RRY 22 person-rem/RRY	U.S. Population Occupational	4.1	GESMO, Tables IV E-8, E-9, and E-12
High-Level Waste:				
Tank Storage	(See Reprocessing)			
Solidification	10 person-rem	Total annual dose to the population within 50 miles of the facility.	4.2.2 and Table 4.2	NUREG-0082, Section V-B
Interim Storage at FRP	(See Reprocessing)			
Transportation	0.011 person-rem 0.046 person-rem 0.041 person-rem	20 transport workers General public: 20 onlookers General public: 7.6×10^5 residents	4.9 Table 4.35	GESMO, Chapter IV, Section G
Interim Storage at RSSF	$< 5 \times 10^{-3}$ person-rem/yr	Population within a 50-mile radius in year 2005.	4.2.5	Letter to W. P. Bishop from J. R. LaRiviere, Oct. 1, 1976, Attachment 4.
Disposal	3.6×10^{-7} rem/RRY-individual 9.0×10^{-4} person-rem/RRY Population	50-year dose commitments (HLW plus TRU)	4.4 and Table 4.18	GESMO, p. IV H-46, and Tables IV H-18 and H-19
Transuranic Wastes:				
Treatment Combustible Noncombustible	0.6 mr/year	At a reprocessing plant	4.3.2.4	BNFP, FES, Reprint 1, p. V-15.

Table 2.11 (Cont'd)

DOSE COMMITMENTS - SUMMARY

(Normal Operations)

	Dose	Basis	NUREG-0116 Section	Reference
Interim Storage	(See Reprocessing)			
Transportation	1.25 person-rem	For 80 transport workers	4.9	GESMO, Chapter IV, Section G
	0.12 person-rem	General Public: 80 onlookers	Table 4.35	
	0.11 person-rem	General Public: 2.7×10^6 residents		
Disposal	3.6×10^{-7} rem/RRY-individual	50-year dose commitments (TRU plus HLW)	4.4 Table 4.18	GESMO, p. IV H-46 and Tables IV H-18 and H-19
	9.0×10^{-4} person-rem/RRY-population			
Long-Term Risks-Disposal				
All HLW accumulated to the year 2000 (commercial)	Table 4.19 50-yr accumulated dose; released in year 2100		4.4.2	BNWL-1927
	Table 4.20 50-yr accumulated dose; released in year 2000			
	Table 4.21 50-yr accumulated dose; released in year 102,000			
	Table 4.22 50-yr accumulated dose; released in yr 1,002,000			
	Table 4.23 50-yr dose commitments for meteorite			ORNL-TM-4639
	Table 4.24 impact at 1,000 and 100,000 years after the year 2000.			
Spent Fuel as a Waste:				
Water Basin Storage	110 person-rem, worldwide - ^{85}Kr cumulative 1975 to 2000		Table 4.25	GESMO, Chapter IV, Table IV K-1
	1.2×10^4 person-rem occupational dose, cumulative 1975 to 2000			
Disposal	260 person-rem	Population dose normalized to model reactor	Table 4.28	GESMO, Page IV E-30 (see paragraph 4.6.3.4)
Low Level Wastes:				
	1% of established guidelines - burning an entire drum of waste.		4.7.3.4	ERDA-1537

Table 2.11 (Cont'd)

DOSE COMMITMENTS - SUMMARY

(NORMAL OPERATIONS)

	Dose	Basis	NUREG-0116 Section	Reference
Transportation	0.61 person-rem	For 75 transport workers	4.9	GESMO, Chapter IV, Sec. G.
	0.14 person-rem	General public: 70 onlookers	Table 4.35	
	0.13 person-rem	General public: 2.4×10^6 residents		
Decommissioning	37 person-rem	Occupational exposure normalized to model LWR fuel requirements	Table 4.31	"Operational Health Physics During Dismantling of the Elk River Reactor," D. McConnon.
Sabotage				
Spent Fuel Storage	20 mrem-whole body at site boundary		4.10.3	NEDM-20682, "Sabotage Analysis for Fuel Storage at Morris".

CHAPTER 3

THE FUEL CYCLE--SOURCE OF WASTES

The nuclear fuel industry is made up of a number of steps, each requiring an operational facility which has its own environmental impacts and in which are generated wastes, not all of which are radioactive.

In GESMO, five alternative fuel-cycle options were evaluated, and three were reviewed in detail: no recycle, U-only recycle, and U/Pu recycle. WASH-1248 had as its base a modification of the U-only recycle process in which separated plutonium was stored for possible later use, rather than being recycled or treated as a waste as in GESMO. This Supplement considers the no-recycle option in addition to the uranium-only recycle option. For the latter, plutonium is treated as a waste (see Sec. 4.5). The uranium-plutonium recycle option is treated in detail in GESMO.

In GESMO the model LWR reactor was an averaged-industry model (2/3 PWRs and 1/3 BWRs) with a fuel reload of about 32 MTU per year for a 1000-MWe plant operating at an 80% capacity factor, with 32.6% efficiency and a fuel burnup of 27,500 MWd/MT. WASH-1248 examined both BWRs and PWRs and their individual effects and maximized impacts for the model LWR by conservatively assuming an average lifetime annual fuel requirement of 35 MT (WASH-1248, pp. S-4 and S-5).

In this Supplement the model fuel-cycle facilities, in terms of capacities, waste generation rates, and types of waste produced, are drawn from GESMO, and the environmental impacts associated with reprocessing and waste management activities are normalized to a model reactor year corresponding to that in WASH-1248, hereafter called "reference reactor year" (RRY). The effects of using the uranium-plutonium recycle option are occasionally outlined for information purposes, but they are not detailed nor considered as part of the amendment to Table S-3.

3.1 Fuel-Cycle Options

This section presents summary descriptions of the three fuel-cycle options from GESMO, including a description of the nature and volume of the wastes from each segment of the fuel cycle.

3.1.1 No Recycle

The no-recycle option is schematically presented in Figure 3.1. Natural uranium is mined in either open-pit or underground mines. The ore is then transferred to mills where it is processed to produce uranium oxide, or "yellowcake." A conversion facility prepares the uranium oxide from the mills for enrichment by converting it to the easily volatilized uranium hexafluoride (UF_6). The enrichment process separates

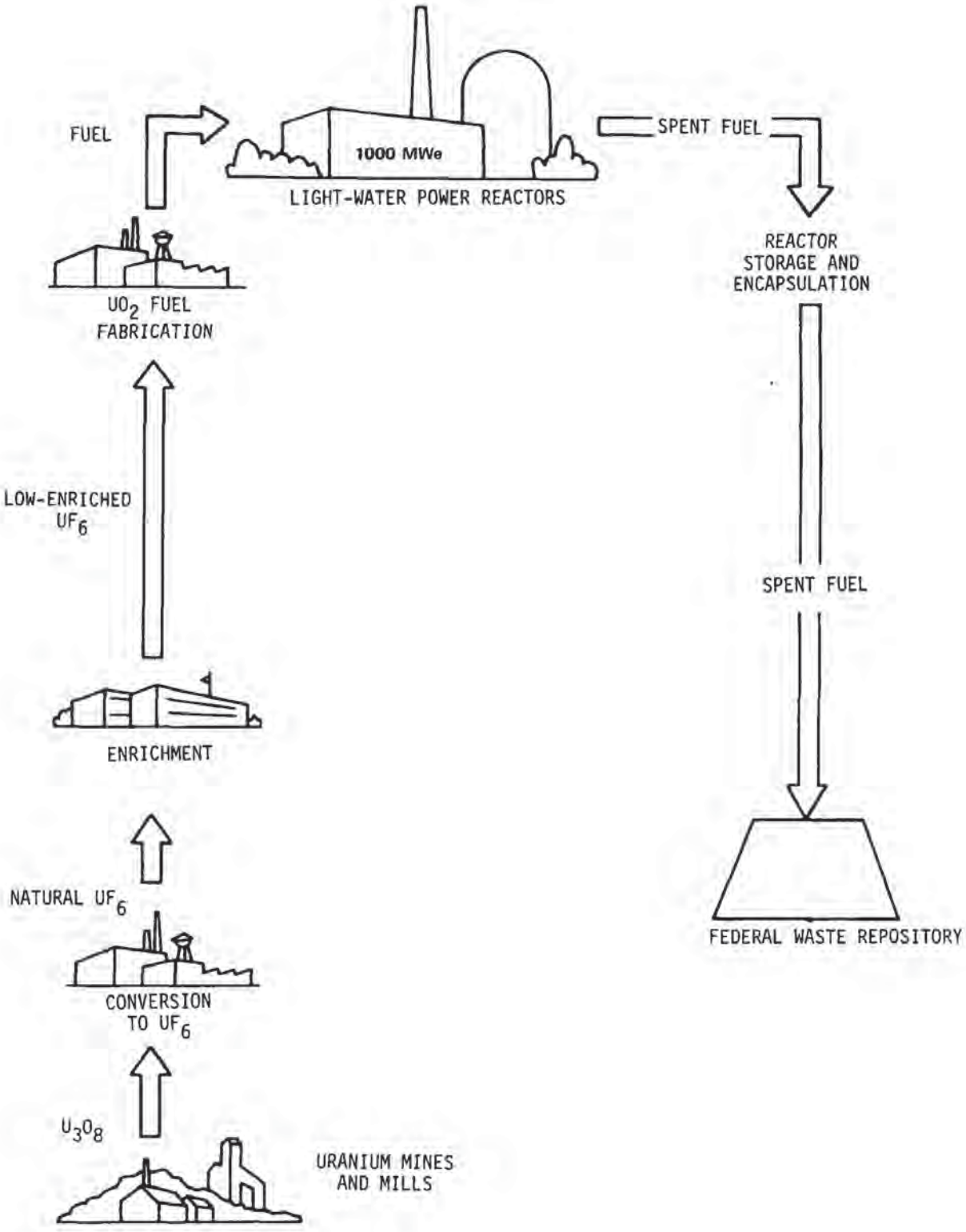


FIGURE 3.1 - NO RECYCLE

the relatively nonfissile isotope U-238 from the more fissile isotope U-235. At a fuel fabrication facility the enriched uranium (~3% U-235) is then converted to UO_2, which is pelletized, sintered, and inserted into tubes to form fuel assemblies. The fuel assemblies are placed in the reactor to produce power. When the content of the U-235 and Pu of the fuel pellet reach a point where the nuclear reactor has become inefficient with respect to neutron economy, the assemblies are withdrawn from the reactor. They are initially stored at the reactor site in large water pools because of their high levels of both heat generation and nuclear radiation.

After storage for sufficient time to allow for short-lived fission product decay and to reduce the heat generation rate, the assemblies would be encapsulated at the reactor and then transferred to a federal repository for interment. Encapsulation may occur at the repository or at an interim storage facility rather than at the reactor. Disposal of spent fuel elements in a repository constitutes the final step in the no-recycle option.

3.1.2 Uranium-Only Recycle

A schematic of the uranium-only recycle option is given in Figure 3.2. The mining, milling, and UF_6 conversion operations are directly comparable to the same operations in the no-recycle option. The principal differences arise in the volumes of materials processed and of wastes generated.

One difference between the no-recycle and U-only recycle options is noted at the enrichment process where the natural UF_6 feed stream is supplemented by recovered, slightly enriched uranium from the reprocessing plant. The enriched and re-enriched UF_6 is then processed to form UO_2 and fuel assemblies as in the previous case. Following irradiation (power production), the assemblies will be stored at the reactor to permit decay of the short-lived isotopes and reduce the heat generation rate. After cooling, the assemblies are transferred to a reprocessing plant for further storage and reprocessing, to recover the residual slightly enriched uranium.

In this option the plutonium is considered a waste material and will be transferred to a federal repository for disposal along with the transuranic (TRU) and high-level wastes (HLW). All these materials will be treated at the reprocessing plant or associated operations to produce stable materials suitable for final disposal. The repository, located in a geologic formation, represents the final stage in this cycle.

3.1.3 Uranium/Plutonium Recycle (Included Here for Comparison Only)

Figure 3.3 is a schematic of the uranium/plutonium (full) recycle option. The processes that are specific to this option are reprocessing and mixed oxide (MOX) fuel fabrication. From the uranium-hexafluoride conversion facility some natural uranium oxide is sent to the MOX fuel fabrication plant for inclusion in Pu-bearing fuel rods. As in the uranium-only case, the slightly enriched uranium recovered in the reprocessing plant is fed into the enrichment plant to supplement the natural

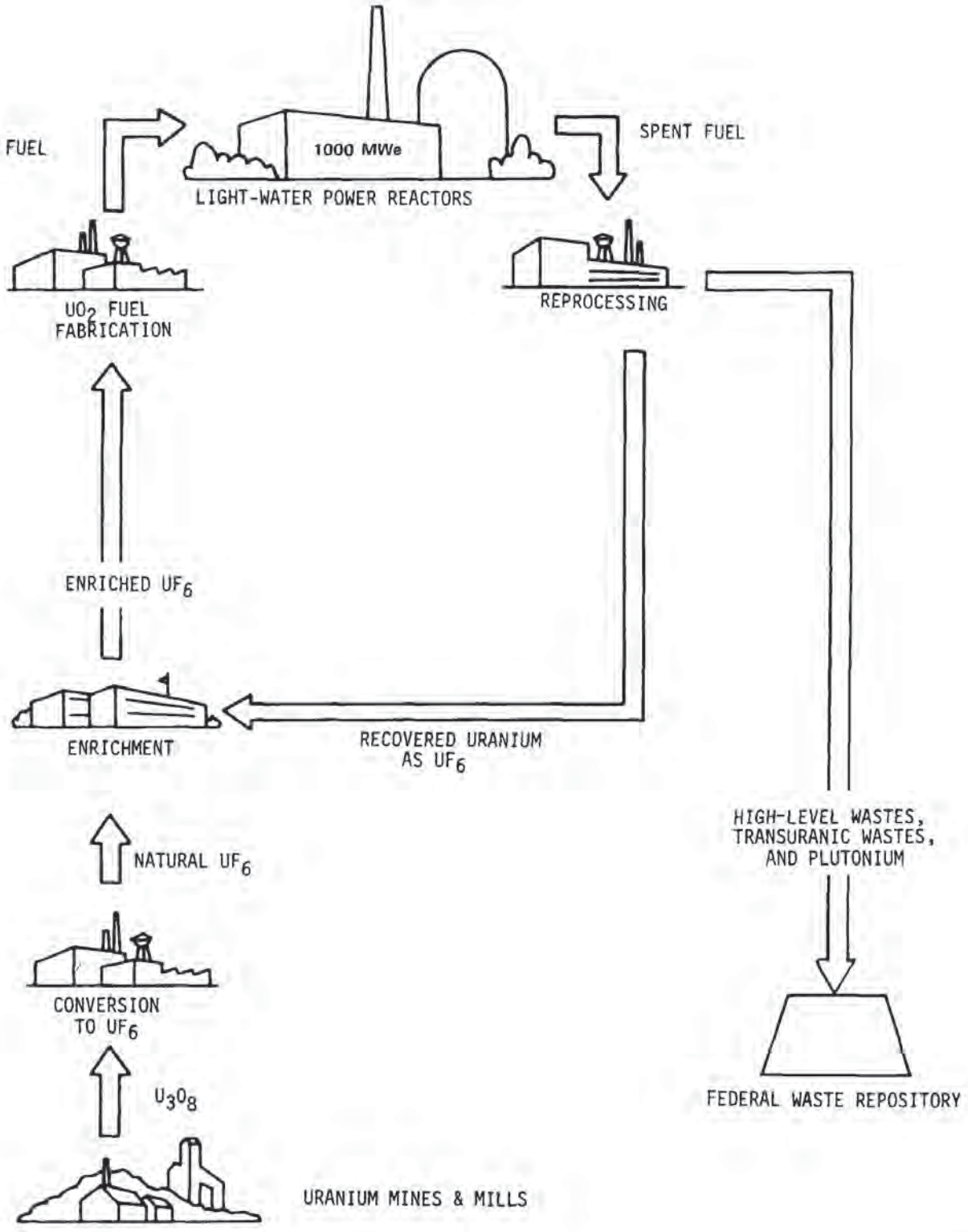


FIGURE 3.2 - URANIUM-ONLY RECYCLE

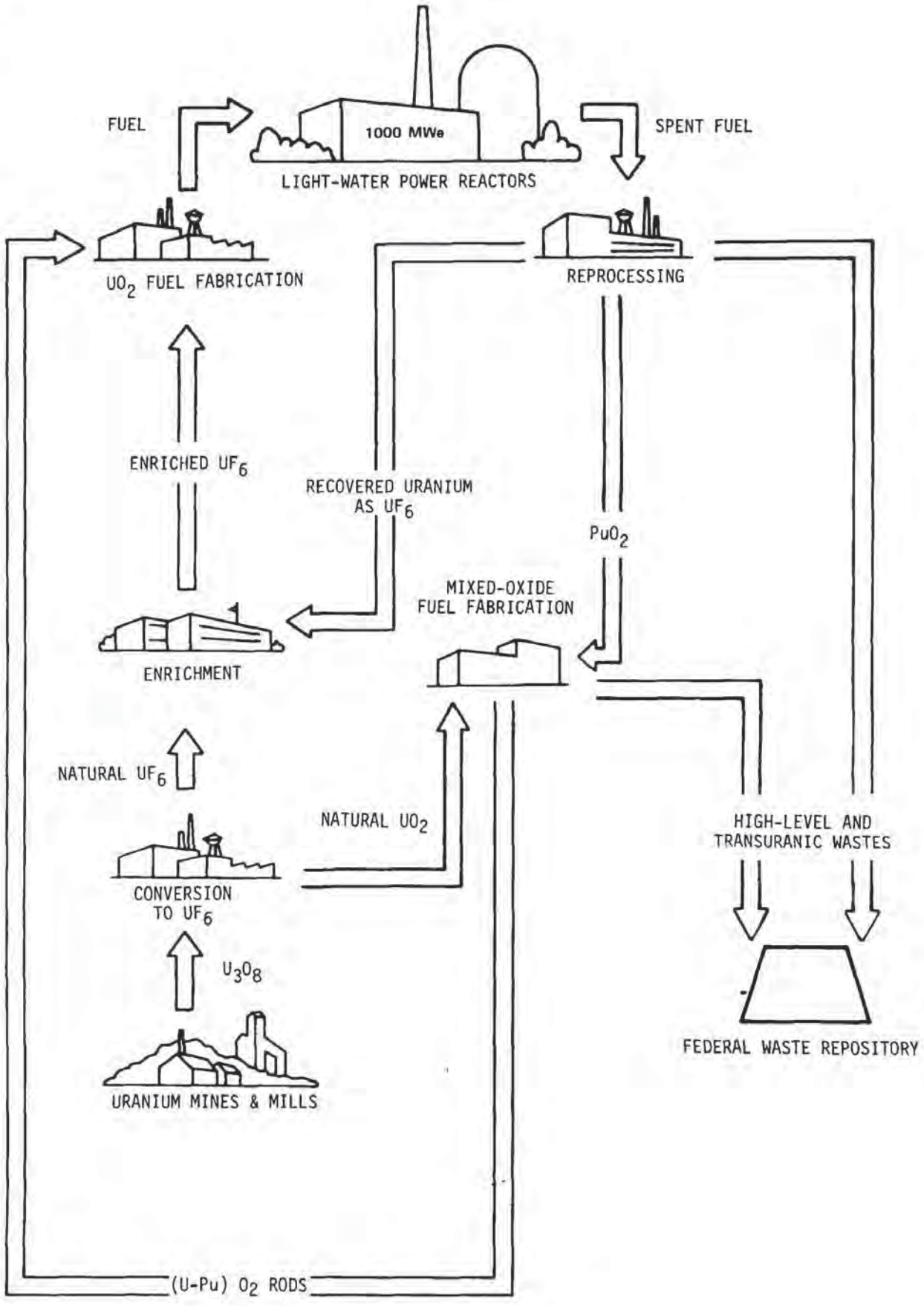


FIGURE 3.3 - URANIUM/PLUTONIUM RECYCLE

uranium from the UF_6 plant. Both the enriched uranium and the MOX fuel streams feed into the fuel fabrication process for the preparation of fuel assemblies for use in reactors. Following removal from the reactor core the spent assemblies are permitted to cool and are then transferred to the reprocessing plant. There the spent fuel is processed to recover slightly enriched uranium and purified plutonium as products, plus several waste streams, including TRU, low-level waste, HLW, and fuel-element hulls. These materials are all treated in the reprocessing plant to obtain solidified materials and are packaged for disposal in a geologic repository or a land burial facility as appropriate.

3.2 Description and Volumes of Wastes from Fuel Cycle Operations

Each facility or operation in the fuel cycle produces wastes. At each facility these wastes are handled, treated, and prepared for shipment if required. As a basis from which to assess the impacts of waste management activities, the volumes and general nature of the wastes are listed here, along with the characteristics of the "model" facility, based on GESMO.

3.2.1 Mining

Uranium ore is obtained from both underground and open-pit mining operations. Wastes from underground mines consist mainly of rock removed in creating shafts and passageways. Wastes from open-pit operations consist largely of overburden removed to expose the ore body. Wastes from either operation are expected to contain only relatively small amounts of uranium. Mine waste should have essentially the background radioactivity typical of the region.

A model mine size has been defined in GESMO (Ch. IV, F) as having an annual production of 18.1 tons of natural U_3O_8 (14.0 MTU) for an underground facility and 181 tons (140 MTU) for an open-pit mine. Little or no waste results from mine operations. Environmental impacts associated with such wastes are dealt with in WASH-1248 and GESMO, and are not further discussed in this Supplement.

3.2.2 Milling

Milling consists of a least four general steps: (1) crushing and/or grinding, (2) chemical dissolution of the ore, (3) physical separation of dissolved ore from the undissolved solid waste, and (4) separation and consolidation of the U_3O_8 product.

Leaching processes at the uranium mill recover more than 90% of the contained uranium but dissolve only a very small fraction of the finely ground ore. The mills now discharge these undissolved solids (mill tailings) along with nearly uranium-free process water and its contained chemicals to an impoundment area (tailings pond) near the mill.

On the basis of assumptions from GESMO (Ch. IV, F and H) of 0.1% U_3O_8 content and 90.5% recovery, about 1.3 metric tons of sands and slimes would be generated

for each kilogram of uranium recovered. Given that the model mill annual capacity is 1050 tons of natural U_3O_8 (807 MTU), the ratio of 1300 for waste to product masses gives an annual waste stream of 1.37×10^6 tons or 525,000 m^3. These wastes are described in detail in GESMO and WASH-1248 and are the subject of a generic environmental impact statement being prepared by the Commission, and are not discussed further in this Supplement.

3.2.3 Hexafluoride Conversion

Before sending natural uranium to the enrichment plant it is necessary to convert U_3O_8 from the mills to readily volatilized UF_6. To produce UF_6, either "wet" or "dry" processes are used. Wastes from UF_6 production facilities contain large quantities of chemicals but very low levels of radioactivity. The wastes are largely composed of CaF_2 ash from dry-process plants, sludge from plants using wet processes, and CaF_2 chemical wastes from treating scrub liquors at both types of plants. The CaF_2 ash is presently drummed and shipped offsite to a burial facility. While low-level sludges may be handled similarly, at present they are stored or buried onsite.

The model UF_6 plant (GESMO, Ch. IV, F and H) using either process has an annual capacity of 15,000 MTU (natural). The wet-process model plant produces 3900 m^3 of wastes while the dry-process model will produce 2480 m^3. In either case, wastes are either disposed of in a licensed burial ground or are buried onsite. Those buried onsite are not discussed further in this Supplement.

3.2.4 Enrichment

The enrichment process raises the isotopic concentration of U-235 in UF_6 feed material from the conversion plants. The annual production rate depends on several factors, including the isotopic concentrations of the feed, tails, and product, and the plant's efficiency. Assuming nominal U-235 concentration of slightly over 3% in the product and 0.3% in the tails, the annual capacity of the model plant is approximately 2,400 MTU (GESMO, Ch. IV, F). To compensate for the presence of U-236 in recycled uranium this figure would be reduced by about 5% in both recycle options. The recycling of plutonium reduces the annual requirement for enriched uranium by about 30% on the average over a reactor's lifetime.

Enrichment is presently performed in plants using the gaseous diffusion process. Wastes generated result primarily from equipment cleanout and uranium recovery. Liquid wastes are impounded in holding ponds where most of the uranium is collected in sludges that are periodically removed and buried onsite. In addition to wastes generated by the enrichment plants, approximately 35 million m^3 of nonradioactive sludges (water and solids) result from operation of the fossil-fueled power plant that supplies power for the enrichment processes. These sludges are disposed of at the power-plant sites.

Future expansions of enrichment capacity are expected to include centrifuge facilities. A gas centrifuge enrichment plant is expected to generate large quantities of nonreusable parts and materials from failed machines. The annual waste from a model centrifuge plant is estimated to be about 56 times greater than from a model diffusion plant of the same capacity. The GESMO model plant (Ch. IV, F) has an annual capacity of 8750 MTSWU and generates 100 m^3 of waste from a diffusion plant and 5600 m^3 from a centrifuge facility. Using a ratio of 4 diffusion plants to 2 centrifuge plants, a waste volume of 1900 m^3 is obtained for an average enrichment plant.

3.2.5 Uranium Fuel Fabrication

A fuel fabrication plant converts enriched UF_6 to UO_2, forms the UO_2 into pellets, loads the pellets into fuel rods, and assembles the loaded rods into fuel assemblies.

The most significant waste generated in these plants is CaF_2, which is formed during the conversion operation at the rate of one metric ton for each metric ton of uranium processed. The uranium content of the CaF_2 is estimated to be about 0.01 μCi/gm. The current practice is to package or store the waste in bulk form onsite. As stated in GESMO (Ch. IV, F), a model fabrication plant will process 1500 MTU annually.

3.2.6 Power Reactors

This Supplement uses the same reactor used in WASH-1248 (pp. S-4 and S-5) to determine fuel-cycle capacity requirements. Wastes generated at the reactor are averaged so as to represent the approximate industry mix of 2/3 PWRs and 1/3 BWRs. The quantity of low-level wastes produced by LWRs is the same for all three fuel-cycle options. Based on a review of reactor licensees' semiannual operating reports through December 1975, a 1000-MWe PWR generates annually approximately 323 m^3 of packaged wet wastes (spent ion-exchange resin, filters, filter sludge, and evaporator bottoms), and 116 m^3 of dry and compacted solid waste. A 1000-MWe BWR produces annually about 850 m^3 of packaged wet wastes and 133 m^3 of dry and compacted solid waste. Using the ratio of 2/3 PWRs and 1/3 BWRs, waste volumes of 500 m^3 of resins and 120 m^3 of dry compacted solids are obtained for the industry-averaged LWRs. An additional 1 m^3 of waste generated in the LWR consists of discarded equipment, in-core instrumentation, and control-rod blades. These wastes contain some 4600 curies of fission products and induced activity.

In the case of the no-recycle option, an additional waste stream, the spent fuel assemblies, is also produced. Given that a PWR fuel assembly is nominally 8.5 inches square and 14 feet long, and assuming that it is overpacked or encapsulated in a 13-inch-diameter casing approximately 16 feet long, each assembly is 15 cu ft in volume. An annual reload for PWR averages 78 assemblies (@ 450 kgU per assembly) giving approximately 1225 cu ft or 35 m^3. Values of 1090 cu ft or 31.2 m^3 are similarly obtained for a BWR, using 200 kg per assembly and 175 assemblies per year; an assembly is 14 cm square by 435 cm long.

The storage of spent fuel, whether at a reactor, an independent spent-fuel storage facility, a reprocessing plant, or a repository, will generate waste resins from water cleanup systems. These are assumed to be constant for all types of fuel and amount annually to 21 m^3 for a 3500-MTHM storage pool. Because the spent fuel in the no-recycle option is stored for 10 years, some 2.1 m^3 of waste are generated per RRY. Less than 1 m^3 is generated for the uranium-only recycle option.

3.2.7 Fuel Reprocessing

In the uranium and uranium/plutonium recycle options the spent fuel discharged from power reactors is reprocessed for its fissile and fertile values. The bulk of the radioactivity from nonvolatile fission products and unrecovered actinides results in the types of wastes described below. The volume of such wastes is essentially independent of which recycle option is used, but the isotopic composition of the radioactive wastes is altered if plutonium recycle is adopted. In the uranium-recycle option, the plutonium becomes a waste to be managed. The model fuel reprocessing plant's annual capacity is 2,000 MTHM (GESMO, Ch. IV, F and H).

3.2.7.1 High-Level Wastes

High-level liquid radioactive waste is defined in Appendix F, 10 CFR Part 50, as "those aqueous wastes resulting from the operation of the first cycle solvent extraction system, or equivalent, and the concentrated wastes from subsequent extraction cycles, or equivalent, in a facility for reprocessing irradiated reactor fuels."

For this analysis the HLW are solidified, cast, formed, or otherwise contained in a primary container which is provided with an overpack or secondary shell as the outer canister. The reference design of the canister (GESMO, Ch. IV, H), including overpack, is a right cylinder some 14 inches in diameter and 10 to 12 feet long (15 ft^3 or approximately 0.3 m^3). It is assumed to hold about 6.3 ft^3 (0.18 m^3) of solidified waste. Since it is estimated that each metric ton of fuel reprocessed from either recycle option will produce about 2 ft^3 of calcine or 2.5 cu ft of vitrified HLW, each canister will hold the waste from 3.14 or 2.52 metric tons of processed fuel, depending on the waste form. Thus 11 or 14 canisters occupying 6 to 8 m^3 will be generated annually from each model reactor.

For the option of fuel recycle of uranium only, the volume of solidified HLW would be essentially the same as above, but about 280 kg of plutonium would require disposal in federal repositories for each model reactor's annual operation, assuming a final Pu concentration of 0.8% in the spent fuel. For purposes of this assessment, it is assumed that following the first cycle of decontamination and the partitioning of the plutonium and uranium streams, the resulting plutonium would be converted to PuO_2 of about 95% purity (5% fission products) and encapsulated. For criticality reasons, the capsules are then supported by spacers in 30-gal containers, each capable of holding 6 kg of the materials* (approximately 47 canisters per year, or 6 m^3 per

*For example, after separation from uranium, the plutonium might not be purified further. Rather, it could be reduced to a solid form and remotely packaged in containers approximately 4 inches O.D. by 24 inches long.

model reactor). It is assumed that the plutonium-bearing canisters will be disposed of in federally operated geologic repositories. The presence of the fission products is considered as a measure which reduces or eliminates the need for safeguards for the plutonium.

Alternatively the solvent extraction process can leave the Pu with the HLW stream so that the waste Pu and fission products may be solidified and managed together. This option, however, raises questions as to criticality and accountability controls.

3.2.7.2 Fuel Element Hulls and Hardware

In the first step of the reprocessing operation, structural components of fuel assemblies are removed and the fuel rods are chopped into short pieces so that the UO_2 or MOX fuel can be leached from them with nitric acid. This operation produces the feed solution for the processing operation and leaves the cladding undissolved. These cladding pieces, called hulls, after additional washing with nitric acid solution, are expected to contain less than 0.1% of the uranium and plutonium present in the fuel (GESMO, Ch. VI, F and H), along with small quantities of fission products and the products of neutron activation of the metal. Approximately 0.5 m^3 of hulls and fuel-assembly hardware result from the processing of each metric ton of heavy metal, regardless of fuel cycle operation, giving approximately 15 m^3 of hulls per reactor on an annual basis. Reprocessing plants as presently designed (GESMO, Ch. IV, F and H), propose interim storage of these wastes in vaults or concrete containers. Because of the residual plutonium contained in these wastes, it is planned that the hulls will eventually be transferred to a federal geologic repository after interim storage at the reprocessing plant.

3.2.7.3 Transuranic Wastes

Reprocessing plants operating under either recycle option will generate plutonium-bearing wastes of several types including laboratory wastes (e.g., small tools and gloves), chemical wastes from cleaning the off-gases, failed equipment, filters, and plutonium extraction solvent, wastes from the plutonium nitrate-plutonium oxide conversion section of the plant, and certain wastes from the process for preparing UF_6 from recovered uranium. The volumes of these wastes (estimated to be nearly 30 m^3 annually per reactor without compaction or incineration) should be nearly independent of the choice of recycle option. The wastes will contain about 0.3% of the plutonium throughput.

The plutonium conversion facilities generate transuranic wastes consisting largely of filters, gloves, discarded process components, and solidified process wastes. It is estimated that these wastes would contain about 0.2% of the plutonium throughput, or approximately 600 grams in 2 m^3.

By using the mean of values estimated for an existing reprocessing plant design, the total annual volume of plutonium-bearing waste generated would approach 44 m^3 per

reactor. The separation facilities would produce minor amounts of solidified aqueous wastes from the solvent and off-gas cleaning systems. These wastes would be expected to contain 300 grams of plutonium with the uranium-only recycle (or about 350 grams of plutonium if both plutonium and uranium are recycled).

Though some wastes may contain transuranic concentrations sufficiently low to permit their burial in a licensed facility, it is conservatively assumed in this study that these relatively low-level wastes will be sent to a federal geologic repository.

3.2.7.4 Chemical Wastes - UF_6 Conversion

An estimated 1100 metric tons of chemical wastes (i.e., spent electrolyte, CaF_2 from treating fluoride scrub liquors, calcine discharges from the uranyl nitrate-to-UO_2 conversion step, and potassium uranate muds) containing about 42 kg of uranium but negligible radioactivity, would be produced by the UF_6 facility at a reprocessing plant per model reactor year (GESMO, Ch. IV, F and H). These wastes will probably be shipped to a licensed commercial facility for burial, though onsite burial at the reprocessing plant may be possible.

3.2.8 Mixed Oxide Fuel Fabrication

The MOX fuel plants, which produce fuel rods containing uranium and plutonium, are incremental operations resulting from plutonium recycle. Thus, radioactive wastes generated at these plants must be considered as an environmental impact attributable to plutonium recycle, and are noted here for completeness only.

The GESMO model plant (Ch. IV, D) has an annual capacity of 360 MT $(U, Pu)O_2$. The solid wastes for disposal total approximately 10,000 ft^3 (290 m^3) and contain 22 kg of PuO_2, or about 0.1% of PuO_2 throughput (GESMO, Table IV D-9, p. IV-D 30). This waste volume is equivalent to 8 m^3/RRY.

3.2.9 Repositories

Two categories of waste repositories are assumed (GESMO, Ch. IV, H). They differ with respect to their locations in the biosphere. One category is surface or near-surface repositories that are used for disposal of low-level wastes whose TRU content is sufficiently low. The second category is repositories in continental geological formations which will be used for containing solidified HLW, plus all other wastes having appreciable concentrations of TRU radionuclides including plutonium, and spent-fuel elements in the no-recycle option.

The use of storage with ready retrieval as a waste-management option is assumed in this assessment only to provide surge space or in-process accommodation, such as the interim storage of spent fuel, hulls and hardware, or HLW for intervals to permit the decay of short-life radionuclides and to reduce heat-generation rates to levels compatible with geologic disposal.

Wastes generated at either surface or subsurface repositories are expected to be small if not negligible, and will be interred at the site of generation.

3.3 Projected Annual Waste Generation

Annual waste quantities generated by fuel-cycle facilities operating at full capacity are given in Table 3.1 for the options of no recycle and uranium recycle. These figures are based on the GESMO model plants as given above and the data are modifications of those given in Table IV H-4, Chapter IV, Section H of GESMO. A change in the GESMO figures for reactor wastes was made to reflect the 2/3 PWR-1/3 BWR mix.

The values in Table 3.2 give the average annual capacity requirements for the individual fuel cycle components to support a 1000-MWe LWR over its lifetime by the alternative fuel cycles. These figures are comparable to those given in Table S-1 of WASH-1248, p. S-5. Minor differences between these figures result from differing assumptions as to tails and fresh fuel assays. These figures are then applied to the data in Table 3.1 and the average annual waste-generation levels are obtained for a model LWR by fuel cycle option. These latter figures are summarized in Table 3.3.

3.4 Cumulative Waste Volumes to the Year 2000

Using those estimates, Table 3.4 is obtained (GESMO, Ch. IV, H), which illustrates possible volumes of accumulated wastes that would come from a balanced LWR power system by the year 2000 for the no-recycle and uranium-only recycle options.

Table 3.1

ANNUAL WASTE PRODUCTION FROM MODEL FUEL-CYCLE FACILITIES

Source	Annual Capacity	Types of Waste	Method of Disposal	Volume of Waste (Cubic Meters)	Radioactivity (Curies)
UF_6 Production					
Dry process	15,000 MTU	Low-level CaF_2; chemical waste	Buried onsite or in licensed burial facility	3,900	350
Wet process	15,000 MTU	Low-level CaF_2; sludges; chemical waste	Buried onsite or in licensed burial facility	2,480	350
Enrichment					
4-Gaseous diffusion 2-Gas centrifuge	8,750 MTSWU	Low-level miscellaneous	Buried onsite or in licensed burial facility	1,900	Not estimated
Fuel Fabrication					
Enriched uranium	1,500 MTU	Low-level CaF_2; miscellaneous	Buried onsite or in licensed burial facility	1,225	Negligible
Mixed oxide	360 MTHM	Plutonium-bearing solids	Federal repository	290	7,200
Reactor Operation	0.8 GWy(e)	Low-level miscellaneous	Buried in licensed burial	620	4,000
		Spent fuel	Federal repository	35	11,000,000*
Spent Fuel Storage	3,500 MTHM	Low-level miscellaneous	Buried in licensed burial	21	Not estimated
Fuel Reprocessing	2,000 MTHM	Low-level miscellaneous	Buried in licensed burial	400	Negligible
		High-level wastes	Federal repository	456	~900,000,000
		Plutonium (30 gallon canisters)	Federal repository	310	~5,000,000
		Miscellaneous transuranium solids (i.e., hulls, hardware, Pu-bearing solids, miscellaneous lab waste)	Federal repository	2,520	~175,000,000

* For 10-year-old spent fuel.

Table 3.2

LIFETIME ANNUAL AVERAGE FUEL-CYCLE REQUIREMENTS FOR A MODEL 1000-MWe LWR BY FUEL-CYCLE OPTION

	No Recycling	Uranium Only
Fresh Fuel Assay (wt% U-235)	3.1	3.2
Spent Fuel Assay (wt% U-235)	0.84	0.84
Ore Supply (0.1% U)(90% Rec.) ($MT \times 10^3$)	272	230
Yellowcake U_3O_8 supply (MT)	293	244
Natural UF_6 (MT)	360	304
Recycle UF_6 (MT)	- -	49
Separative work (MTSWU)	127	134
Enriched UF_6 (MT)	52	52
Enriched UO_2 (MT)	40	40
Fuel Loading (MTU)	35	35

Bases

Reactor plant load factor - 80%

Enrichment tails assay - 0.3%

Losses of 1% each in fuel fabrication and reprocessing

Fuel burnup 33,000 MWd/MTHM

MT = metric ton = 2,205 lbs

MTU = metric ton uranium

SWU = separative work units

Table 3.3

ANNUAL WASTE VOLUMES AND RADIOACTIVITY FROM FUEL-CYCLE OPERATIONS BY FUEL-CYCLE OPTIONS, AVERAGED OVER THE LIFETIME OF A 1,000-MWe LWR

Fuel Cycle Operation	Annual Capacity	Waste Type	Waste Volume Per RRY (Cubic Meters)		Radioactivity (Curies)	
			No Recycle	Uranium Only	No Recycle	Uranium Only
UF_6 conv. (dry)	15,000 MTU	CaF_2 Chem Waste	92	95	6	5
(wet)	15,000 MTU	CaF_2, Sludge Chem Waste	41	35	6	5
Enrichment	8,750 MTSWU	Low-Level Misc.	28	30	Not estimated	
Fuel Fabrication						
U	1,500 MTU	CaF_2, Misc.	29	29	Negligible	
Reactor	1,000 MWe @ 80% capacity factor	Low-Level	620	620	4,600	4,600
		Spent Fuel	35	--	11,000,000*	---
Spent Fuel Storage	3,500 MTHM	Low-Level	<3	<1	Not estimated	
Fuel Reprocessing	2,000 MTHM	Low-Level Misc.	--	7	--	Negligible
		High-Level Waste	--	8	--	~16,000,000
		Misc. TRU	--	44	--	~3,000,000
		Plutonium	--	6	--	~90,000

*For 10-year-old spent fuel

Table 3.4

CUMULATIVE VOLUMES OF WASTE INVENTORY IN THE YEAR 2,000 (m^3)

Type of Waste	Fuel Cycle Option	
	No Recycle	U Recycle
Mill Tailings	7.8×10^8	6.9×10^8
Spent Fuel	55,000[a]	6,000[c]
High-Level	b	6,500[d]
Transuranic	b	76,500[e]
Hulls and Hardware (Transuranic)	b	52,000
Low-Level Reactor Waste (Nontransuranic)	3.8×10^6	3.8×10^6
Other Low-Level (Nontransuranic)	310,000	300,000
Chemical	179,000	183,000

[a] 400,000 spent-fuel assemblies.

[b] Not produced with no-recycle.

[c] 37,000 spent fuel assemblies in pool storage awaiting processing.

[d] Volume of HLW in 37,000 canisters.

[e] Includes plutonium wastes.

CHAPTER 4

DISCUSSION OF IMPACTS OF REPROCESSING AND WASTE MANAGEMENT

Introduction

For discussion, the steps in the reprocessing and waste management parts of the fuel cycle have been divided into individual operations, each with a discernible set of environmental impacts. For each of these steps a technology has been selected for discussion, and is described and discussed relative to the environmental impacts of its operation.

Each section begins with an introduction which defines the waste form, cites management constraints, describes the reference waste-management system, and presents information and assumptions common to all subsections.

Each subsection in this chapter is organized as follows:

Technology Selection: a brief discussion of alternative technologies and the reasons for selection of the reference technology;

Description: elaborates on the reference technology, describing what it does (objective), and how it does it (process);

Bases for Impacts: reference environmental or site characteristics; analytical considerations for normal and accident conditions, and mitigating techniques; and,

Environmental Impacts, which include:

- Land use - The commitment of land for a facility site can be either "permanent" or "temporary." A temporary commitment is for the life of a plant or succeeding plants, after which the land may revert to any desired use. A permanent commitment refers to land that may not be released after plant shutdown or decommissioning.

- Water use - Water is used in reprocessing and waste management activities as a coolant and for process requirements. Environmental considerations are based upon the way water is returned to the biosphere.

- Fossil fuel use - The major resource use (aside from land and water) is other energy requirements for operation of the facilities.

- Nonradiological effluents - Gaseous, liquid, and solid materials which are released to the environment are estimated.

- Radiological effluents - All releases of radioactive nuclides are estimated and reported.

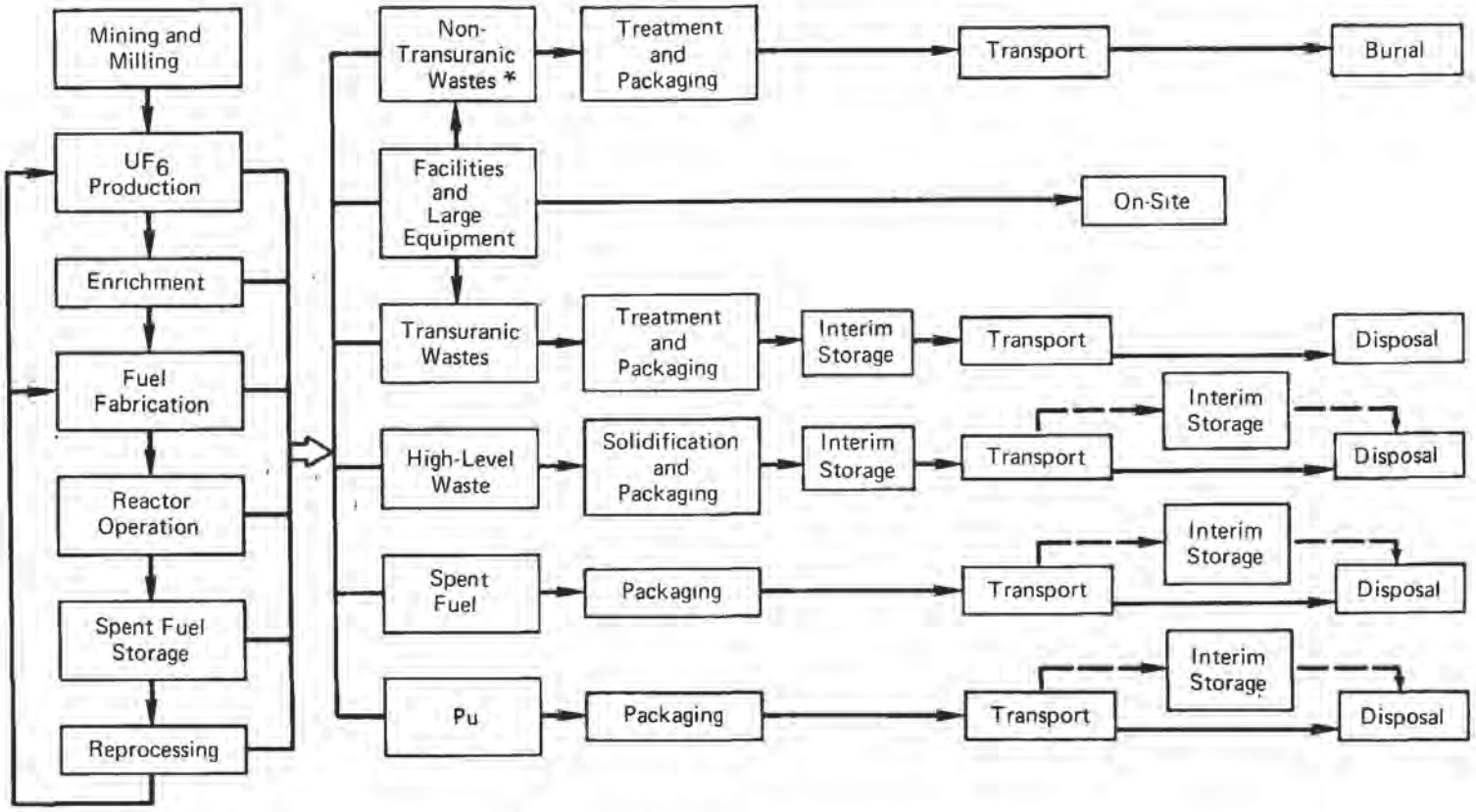
- Buried solids (radioactive) - Although not released to the biosphere, solid wastes buried at the site of the operation are calculated and reported.

Where data were available, doses resulting from radionuclide releases from reprocessing and waste management are given.

Reference System

The reference waste management system for the uranium-only recycle option is as follows: spent fuel is reprocessed in a 2000 MT/yr facility employing the Purex solvent extraction process (Sec. 4.1). HLW will be stored as a liquid (Sec. 4.2.1) for up to 5 years at the FRP in 300,000-gallon stainless-steel tanks similar to the BNFP design; the waste will be solidified (Sec. 4.2.2) at the FRP using the spray calciner/in-can melting process; the reference waste form is glass. Solidified waste will be packaged in stainless-steel cylinders. The volume of waste contained in each canister is about 6.3 cubic feet (see Fig. 4.2). Waste canisters will be retrievably stored (Sec. 4.2.3) in a water basin at the FRP for up to 10 years. Transportation of HLW canisters (Sec. 4.2.4) will be by rail. Structural and containment features of transportation casks will be similar to those of casks for irradiated fuel. For the sake of completeness, the environmental analysis of an interim storage facility (RSSF) is included (Sec. 4.2.5). The reference RSSF is based on the sealed storage cask concept. HLW disposal (Sec. 4.2.6) is assumed to be in bedded salt at a federally managed facility. TRU wastes will be treated (Secs. 4.3.1 and 4.3.2), retrievably stored at the site of generation on earth-covered surface pads (Sec. 4.3.3), and transported (Sec. 4.3.4) to a federal repository for disposal (Sec. 4.3.5). The federal repository will provide for disposal of all HLW and TRU waste in a bedded-salt formation by conventional mined emplacement (Sec. 4.4). Low-level wastes will be treated and packaged (Sec. 4.7.1) using established technology and procedures, and disposed of by shallow land burial (Sec. 4.7.3). In the uranium-only recycle option, the plutonium will be handled, treated (Secs. 4.5.2 and 4.5.3), and disposed of as a waste (Sec. 4.5.5). The option of storing plutonium for future use is not considered. In the no-recycle option, the spent fuel is disposed of as a waste (Sec. 4.6).

To assist the reader in following each waste management system, flow diagrams like the one below appear at the beginning of each section, highlighted to show the subject of that section. Nuclear data for radionuclides of interest are given in Appendix A.



* Low-Level Wastes

4.1 Reprocessing

Reprocessing refers to operations performed on spent reactor fuel to separate its reusable products from its wastes. The fuel used in LWRs consists of assemblies of long, sealed zircaloy tubes filled with small ceramic pellets of UO_2. During the life of the fuel in the reactor, fission and irradiation products are formed in the ceramic. The spent fuel contains residual U-235 (part of the original fuel), plus plutonium formed by the transmutation of U-238. Both of these are useful as fissile material. Chemical processes are used at the fuel reprocessing plant (FRP) to separate these fissile products from fission-product wastes and from each other. For regulatory and economic reasons neither the product nor the waste solutions can be shipped. Therefore, the reprocessing plant also includes facilities to convert the uranium product to UF_6, the plutonium product to PuO_2, and to solidify the liquid wastes that remain.

The evaluation of environmental impacts associated with fuel reprocessing discussed in this section is based upon GESMO.[1*]

4.1.1 Technology Selection

The Purex solvent extraction process, which has been the process of choice for fuel reprocessing for many years, has been selected for evaluation in this Supplement. This process, developed by the AEC, has been in use at ERDA-AEC plants since about 1954 for reprocessing natural and low-enrichment fuels. It has been used at reprocessing facilities in other countries as well.

The first reprocessing system used in this country was the bismuth phosphate process, a gravimetric chemical process for recovering plutonium from irradiated natural uranium, and which produced a voluminous neutralized waste.[2] Solvent extraction processes were also considered during development of the bismuth phosphate process, but were not considered to be as certain of success. Research and development on solvent extraction for uranium and plutonium recovery continued and several plants began successful operation at AEC (now ERDA) facilities in the early 1950's. The first solvent extraction process to operate for reprocessing of natural uranium fuels was the Redox process[3] which also produced a large volume of waste because of its use of aluminum nitrate as a salting agent. Continued development in solvent extraction resulted in the metal recovery and Purex processes. The metal recovery process was used to recover uranium from bismuth phosphate waste, while Purex process plants were constructed at the Hanford and Savannah River plants for processing of irradiated natural and low-enriched uranium. Purex process plants have operated successfully for over 20 years in these government-owned facilities. Since the Purex process uses nitric acid as the salting agent and produces a much smaller volume of waste than the Redox system, it is the only process now in use for natural or low-enriched fuels at ERDA plants. Similar solvent extraction processes have been developed and used for the reprocessing of highly enriched fuels at other ERDA facilities such as the Idaho Chemical Processing Plant.[4]

*References are listed at the end of Chapter 4.

4.1.2 Description

Fuel assemblies are stored for at least 150 days after removal from the reactor to allow for the decay of the short-half-life radionuclides before reprocessing. All planned commercial plants use mechanical means for preparing the fuel assemblies for dissolution. Chemical systems for recovering special nuclear material are adaptations of the Purex process, for which the technology is well defined.[4] However, specific methods for mechanical disassembly of fuel assemblies, dissolution of fuel material, and recovery of uranium and plutonium differ somewhat from plant to plant.

The Purex process uses an organic solvent consisting of 30% tri-n-butyl phosphate in a hydrocarbon diluent. The irradiated fuel is sheared or otherwise disassembled and the exposed fuel material dissolved in hot nitric acid, leaving behind the chopped tubing (hulls). The nitric acid solution containing the dissolved fuel material is processed through a series of solvent extraction cycles. An aqueous nitric acid effluent stream from this process contains the fission products from the fuel and constitutes the bulk of the HLW as defined by 10 CFR Part 50, Appendix F. The uranium and plutonium are separated and further purified in preparation for conversion to uranium hexafluoride and plutonium oxide.

4.1.3 Bases for Impacts (Model Fuel Reprocessing Plant)

Three commercial nuclear fuel reprocessing plants have been constructed in the United States, each on a relatively large, remote site:

Table 4.1

IRRADIATED FUEL REPROCESSING PLANTS
Site Data and Demography

Plant and Location	Plant Capacity MTU/day*	Site Size, Acres	Major Nearby Cities	Population	Distance Miles
Nuclear Fuel Services West Valley, N.Y.	1**	3,500	Springville, N.Y. Olean, N.Y. Buffalo, N.Y.	4,350 19,000 463,000	4.5 25 28
Morris Operation*** Morris, Ill. (General Electric)	1	890†	Morris, Ill. Joliet, Ill. Aurora, Ill.	8,000 79,000 74,000	8 15 27
Barnwell Nuclear Fuel Plant Barnwell, S.C. (Allied-General Nuclear Services)	5	1,700††	Barnwell, S.C. Aiken, S.C. Augusta, Ga.	4,500 16,000 60,000	7.5 26 33

*Metric tons of uranium per day.

**NFS has applied for a license to operate at 2.5 MTU/day.

***Formerly called the Midwest Fuel Recovery Plant, now used to store spent fuel.

†Adjacent to the Dresden nuclear reactor site of 2,230 acres.

††Adjacent to ERDA's Savannah River Plant exclusion area.

All of these commercial FRPs are based on the Purex process. The BNFP has a larger capacity than either NFS or MFRP. The NFS plant operated from 1966 to early 1972, at which time it was shut down for proposed modifications to increase capacity to 750 tons per year. On September 22, 1976, NFS notified NRC that they are abandoning plans to reprocess fuel because, based on their current capacity estimate of 600 tons per year, it could not be an economically competitive operation.

The MFRP was designed to use only one stage of solvent extraction decontamination, relying on the UF_6 conversion process to complete uranium decontamination. Therefore, this plant required operation of uranyl nitrate calcination and UO_3 fluorination equipment under remote conditions in a shielded canyon. All previous plants employing these operations have used natural or decontaminated uranium in these processes, permitting hands-on maintenance and operation. After months of testing at MFRP, General Electric concluded that the facility could not be operated successfully unless conventional uranium decontamination steps were added, but has made no decision to so modify the plant.

The BNFP fuel reprocessing and UF_6 facilities are finished and in preoperational testing, but licensing for BNFP is still in process.

Figure 4.1 is a simplified flow diagram of the major operations performed at the spent-fuel reprocessing plant complex. The fission products are separated from the uranium and plutonium, and the latter are recovered as purified nitrate solutions in the separation facility.

The plutonium nitrate solution may be purified and converted to solid plutonium dioxide (PuO_2), which may be blended with UO_2 for recycling as LWR mixed-oxide (MOX) fuel elements or shipped as waste to a federal repository. Pursuant to 10 CFR 71.42, after June 17, 1978, plutonium shipments must be in the form of a doubly confined solid. This is also the preferred form for the storage of large inventories of recovered plutonium.

The uranyl nitrate is purified and converted to uranium hexafluoride (UF_6), which is returned to the fuel cycle as feed for uranium enrichment plants.

The radioactive HLW which contain most of the fission products and transuranic (TRU) elements that were separated from the uranium and plutonium may be confined initially in high-integrity tanks within vaults before being converted to encapsulated immobile solids. Pursuant to Appendix F of 10 CFR Part 50, liquid HLW must be converted to solids within 5 years after generation. Within 10 years after generation these solid wastes must be transferred to a federal repository for storage.

In addition to the HLW, a variety of other contaminated wastes results from operations of the separation facility, the UF_6 conversion facility, the PuO_2 conversion facility, and waste handling. Most of these wastes may be contaminated by long-lived fission products, TRU radionuclides, or both. See Sections 4.2 to 4.4 and 4.7 for the treatment and disposal of these wastes.

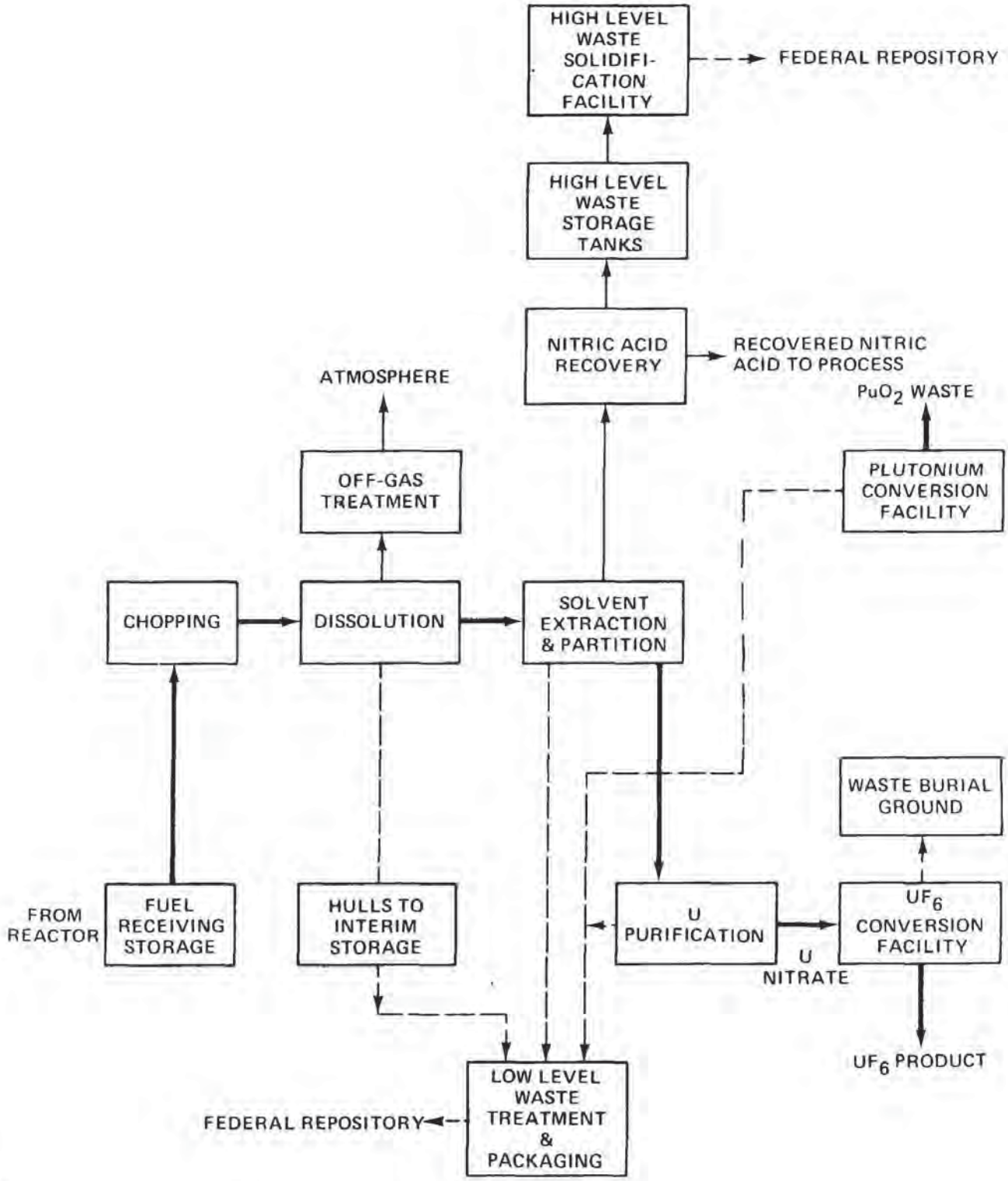


FIGURE 4.1 - FLOW DIAGRAM OF REPROCESSING PLANT
(URANIUM-ONLY RECYCLE)

The Commission is considering amendment of its regulations in 10 CFR Part 20, "Standards for Protection Against Radiation," to prohibit the disposal by burial in soil of waste containing significant quantities of transuranic elements per gram of waste.[5] The proposed amendment would require that waste materials containing TRU elements be transferred to a federal repository as soon as practicable, but within 5 years after generation. If this amendment is adopted, essentially all radioactive waste materials from reprocessing plants, except possibly some solid waste associated with fuel receiving and storage operations and the UF_6 conversion process, would have to be handled as TRU-contaminated waste, requiring transfer to federal repositories.

Section IV E-1.2 of GESMO presents the rationale and analysis for estimating the fuel available for reprocessing for the next 25 years. A model fuel reprocessing plant (FRP) was selected there and is used here. Existing and proposed FRPs have design capacities from 300 to 2100 MT/yr. Although future plants are likely to have larger capacity (over 3000 MT/yr) than those currently constructed, the model FRP for this evaluation is taken to be similar to current designs, located on a 2000-acre site in a sparsely populated area, and with a design throughput capacity of 2000 metric tons of heavy metal (U and Pu) per year.

Structures, systems, and components that confine radioactive materials are designed to withstand the effects of earthquakes and other natural phenomena such as tornadoes, hurricanes, and floods. Cooling water from wells or a surface stream is returned to a surface stream via cooling towers and cooling ponds. All process and ventilation air that might contain radioactive contaminants is treated or filtered, or both, before discharge to the atmosphere via a stack 100 meters tall. The plant has standby diesel equipment for emergency power and water supplies, an emergency water distribution system, water treatment systems, boilers to produce steam for heating and process needs, air compressors, electrical switchgear, and sanitary waste treatment systems.

4.1.4 Environmental Impacts of the Fuel Reprocessing Plant

A detailed analysis of the model plant operation and its environmental impact is given in GESMO.[6] The uranium-recycle-only analysis is summarized below.

The estimated environmental impact of a 2000-MT/yr FRP that reprocesses an annual average fuel load composed of UO_2 fuel (irradiated to 33,000 MWd/MT and cooled 160 days before reprocessing) is tabulated in Table 4.2. This table presents impact data for all components of the reprocessing complex, including fuel receiving and storage, separation, UF_6 conversion, plutonium conversion, and waste solidification facilities.

In addition to the environmental impact analysis of the GESMO model plant, the NRC staff has published an environmental impact statement for a specific FRP: the Barnwell Nuclear Fuel Plant.[7,8] Analysis of the BNFP was based on the reprocessing of UO_2 fuel only.

Table 4.2

SUMMARY OF ENVIRONMENTAL CONSIDERATIONS RELATED TO REPROCESSING (NORMAL OPERATION)
2000 MT/Yr Model Plant Normalized to RRY

Natural Resource Use	Uranium Recycle
Land (acres)	
Temporarily committed	32
Permanently committed	0.12
Total	32.1
Water (millions of gallons/yr)	
Discharged to air	6.6
Discharged to water	54.8
Total	61.4
Fossil Fuel (per year)	
Electrical Energy (MW-hr x 10^3)	4.0
Equivalent Coal (MT x 10^3)	1.5
*Natural Gas Equivalent, 10^6 cu ft	28.6
Effluents	
Chemical (MT/yr)	
***Gases**	
SO_x	5.4
NO_x	21.9
Hydrocarbons	0.5
CO	0.5
Particulates	0.6
Fluorides	0.05
Liquids	
$SO_4^=$	< 0.02
NO_3^-	---
Cl^-	0.09
Na^+, K^+	< 0.02
Fe	---
Thermal (Btu/yr x 10^9)	
Fossil Fuel	47.4
Decay Heat	28.1
Total	75.5
Radiological (Ci/yr)	
Gases (including entrained matter)	
^3H	18,000
^{14}C	24
^{85}Kr	400,000
^{90}Sr	0.01
^{106}Ru	0.14
^{129}I	0.03
^{131}I	0.83

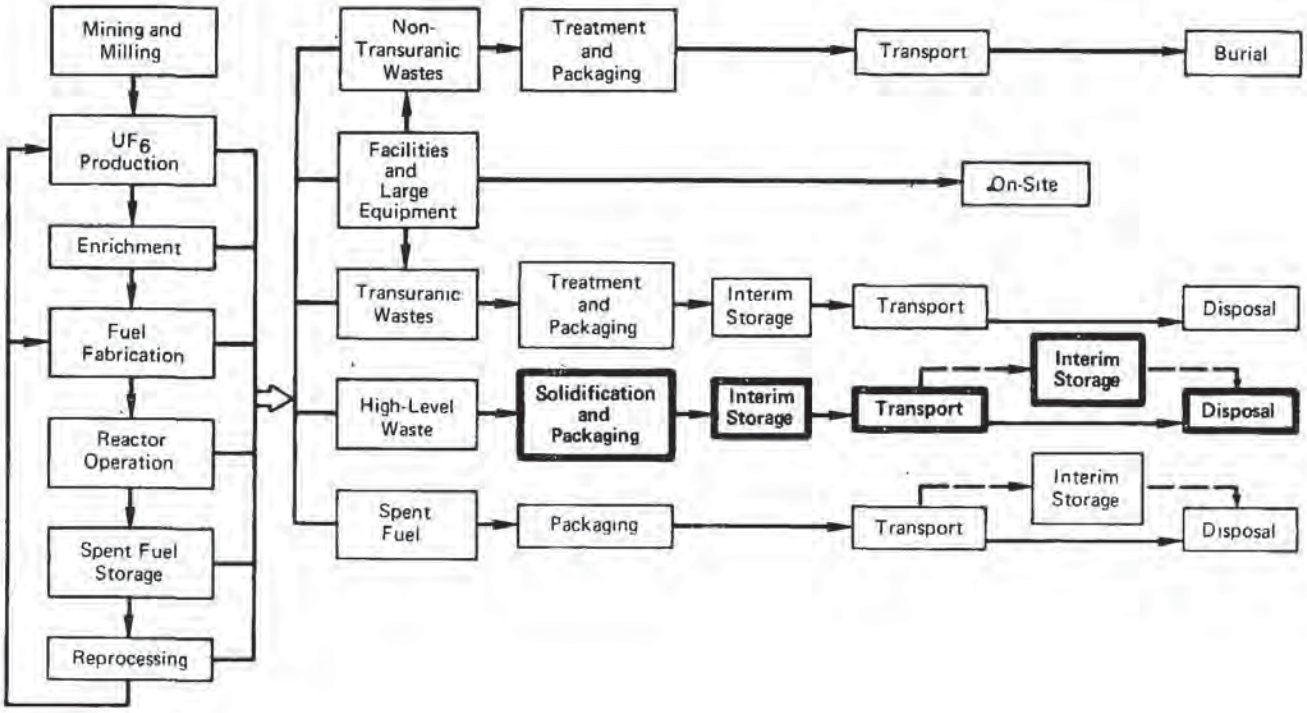
*Thermal equivalent of fuel used at reprocessing plant site. Plant uses 200,000 gal #2 fuel oil and 2,100 gal propane per RRY. Combustion products of fuel oil and propane are included with process effluents in gaseous effluents.

Table 4.2 (Cont'd)

Radiological (Ci/yr) (Cont'd)	Uranium Recycle
Other fission products	0.18
U	3.9×10^{-5}
Puα	8.8×10^{-4}
Puβ	.02
^{241}Am	1.4×10^{-5}
^{243}Am	1.1×10^{-6}
^{242}Cm	1.5×10^{-3}
^{244}Cm	1.1×10^{-4}
Total Fission Products and Transuranics	< 0.2
Population Dose Commitments (person-rem)	
Occupational	22
U.S. Population	330
Foreign Population	210
Total	562

The reprocessing of MOX fuel in addition to UO_2 fuel produces no significant changes in reprocessing throughput, chemical effluents, or power requirements. However, there may be a slight change in isotopic composition of the TRU radionuclides released in the off-gas effluent from the plant when reprocessing MOX fuel. It is assumed here that radioactive effluents will vary directly with throughput and the plant's annual average feed mix. The introduction of plutonium recycle fuel is not expected to cause any significant change in chemical effluents from the plant. The average annual concentration of chemical pollutants (NO_x, SO_x, CO, F) at the plant boundary is not expected to exceed federal or state standards.[7,8]

The calculated environmental impacts from GESMO (Table IV E-8) have been normalized to a reference reactor year (see Ch. 3). This normalization is based on an assumed annual reload of 35 MTU, the use of which is conservative in that it tends to attribute a larger fraction of the environmental impacts to each individual reactor than is actually the case. The normalized impacts from GESMO for uranium-recycle-only are listed in Table 4.2.



4.2 Management of High-Level Waste

High-level liquid radioactive waste is defined in Appendix F, 10 CFR Part 50, as aqueous wastes from the operation of the first-cycle solvent extraction system, or equivalent, and the concentrated wastes from subsequent extraction cycles, or equivalent, in a facility for reprocessing irradiated reactor fuels. Only the fuel cycles involving reprocessing generate HLW under this definition.

The HLW will contain essentially all of the nonvolatile fission products and transuranic elements, and about 0.5% of the uranium and plutonium that was initially in the spent fuel. HLW from 5-year-old spent fuel will contain some 435,000 Ci of fission-product activity and 2,000 to 12,000 Ci* of actinide activity per metric ton of fuel.

Of the many waste types produced in the U-recycle fuel cycle, HLW, although a small fraction of the total waste volumes, has the highest specific activity, requires the most shielding, and presents the greatest need for heat dissipation. These properties, and the potential hazard of the waste, decrease with time. Hazards as a function of time have been calculated by a number of authors (see for example Refs. 1 and 2) and are shown in Figures 4.9 through 4.11. In brief, the significant hazard diminishes with fission-product decay over a few hundred years, while the potential hazard from heavier elements diminishes more slowly.

*The quantity of heat and radioactivity per metric ton of fuel depends upon whether uranium alone, or both uranium and plutonium, are recycled.

Reference HLW System

The following sections are organized sequentially from tank storage of liquid HLW at the reprocessing plant to final disposal in bedded salt. The steps assessed are: tank storage (Sec. 4.2.1); solidification and packaging (4.2.2); interim storage at the reprocessing plant (4.2.3); transportation (4.2.4); interim storage at an RSSF (Retrievable Surface Storage Facility) (4.2.5); and final disposal (4.2.6).

A common fission-product source term is used throughout the HLW section (see Sec. 3.2.7.1), however, this Supplement assumes a conservative case for each fuel cycle segment independent of the other segments, which results in some inconsistencies. For example: (1) although borosilicate glass is the reference waste form, a calcine waste form is assumed for accident analysis at the interim storage facilities, and (2) the solidification section (4.2.2) assumes waste solidification at 189 days, while the tank storage section assumes liquid storage for up to five years, and interim storage at the FRP is assumed to be up to 10 years. The gaseous effluents from these operations are routed to the gaseous treatment system of the FRP; they are considered in Section 4.1.

The reference HLW system is as follows: HLW will be stored as a liquid (Sec. 4.2.1) for up to 5 years at the FRP in 300,000-gallon stainless-steel tanks similar to the BNFP design; the waste will be solidified (Sec. 4.2.2) at the FRP using the spray calciner/in-can melting process; the reference waste form is glass. Solidified waste will be packaged in a 304L stainless-steel cylinder 12-3/4 inches in O.D. by 10-12 feet long. The volume of waste contained in each such canister is about 6.3 cubic feet (see Fig. 4.2). Waste canisters will be retrievably stored (Sec. 4.2.3) in a water basin at the FRP for up to 10 years. Transportation of HLW canisters (Sec. 4.2.4) will be by rail. The structural and containment features of transportation casks will be similar to those of casks for irradiated fuel. For the sake of completeness, the environmental analysis of an interim storage facility (RSSF) is included (Sec. 4.2.5). The reference RSSF is based on the sealed storage cask concept. HLW disposal (Sec. 4.2.6) is assumed to be in bedded salt at a federally managed facility.

SIZE

DIAMETER: 6 TO 24 INCHES

LENGTH: 10 TO 15 FEET

(12-INCH DIAMETER X 10-FOOT LENGTH TYPICAL)

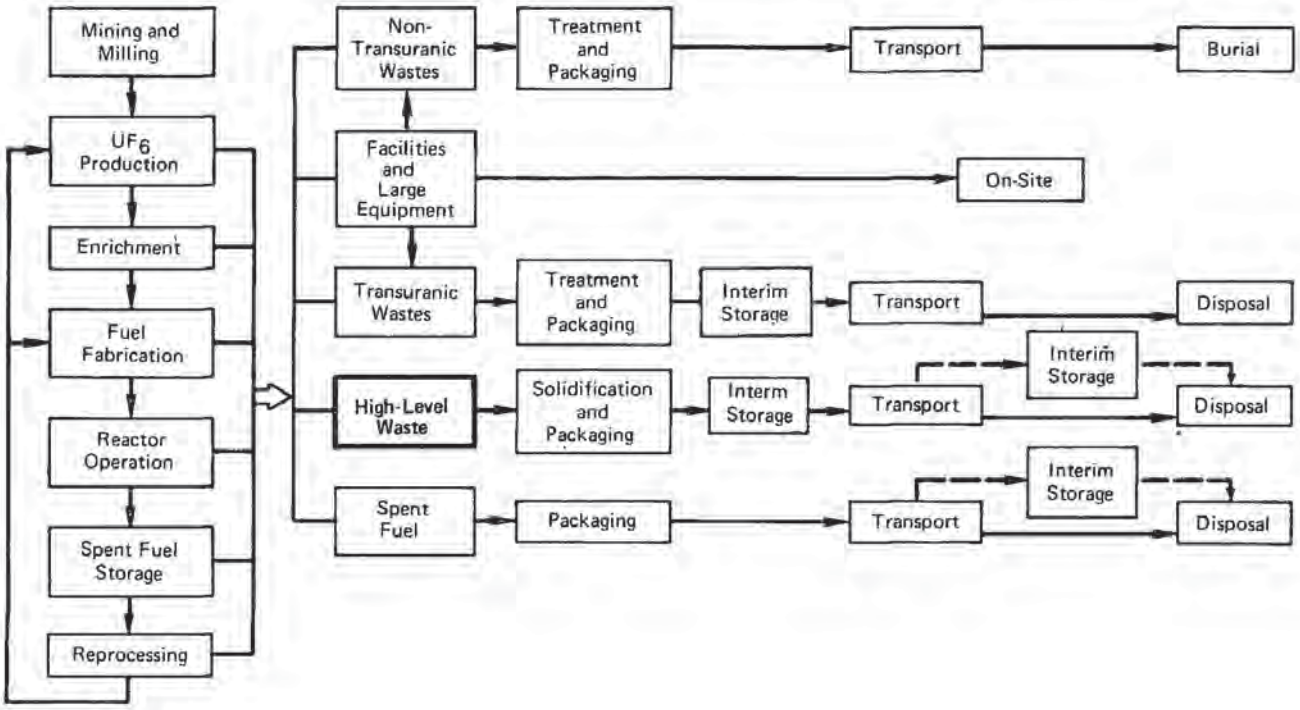
MATERIAL

300 SERIES STAINLESS STEEL

HEAT

1 TO 20 KW (5 KW TYPICAL)

FIGURE 4.2 - HIGH-LEVEL WASTE CONTAINER FOR RETRIEVABLE SURFACE STORAGE FACILITY (RSSF)



4.2.1 Tank Storage

Storage tanks for HLW have undergone a considerable evolution in the 30 years since they were first put in service during World War II at Hanford. The history of tank development at Hanford, Savannah River, and Idaho Falls is given in Appendix E. Knowledge gained at these installations is reflected in tank design at commercial plants that have operated or plan shortly to operate.

NFS Tanks

The NFS West Valley plant was designed to use both the standard Purex and the Thorex processes.[3] Wastes generated by both processes are now in storage at West Valley. There are two storage tanks for neutralized Purex waste at West Valley, one serving as a spare. These are similar to tanks at the Savannah River plant in that the primary tank, including internal support columns, sits in a secondary pan 4 feet high relative to the bottom of the primary tank. Each primary tank has a working volume of 600,000 gallons; one tank is full. The tanks are fabricated of carbon steel and were stress-relieved in the field to minimize stress corrosion. They are equipped with air sparger-circulators for mixing and for purging of radiolytically generated hydrogen. The pans are equipped with level indicators and connected to pumps for transfer of any leakage back to the storage tank or to the low-level liquid waste treatment facility.

Because the neutralized waste in storage at NFS is not sufficiently concentrated to maintain self-boiling, a temporary heat exchanger has been installed in the active-waste storage tank to concentrate the dilute waste. Off-gases and steam

from the tank are passed through a condenser and the condensate is either transferred back to the tank or treated by ion exchange for removal of cesium and strontium before discharge to the low-level waste treatment facility. The off-gas is then heated to 180°F and filtered before discharge.

Thorium waste was generated in processing fuel from Core A of the Indian Point 1 Reactor. The 12,000 gallons of waste[4] are stored in a 304L stainless-steel tank 12 feet in diameter and 15 ft. 9 in. high. The active tank and an identical spare are installed in the same concrete vault, which has a stainless-steel bottom liner and a stainless-steel sump. The tanks are equipped with cooling coils to maintain waste temperature below 140°F. The vault sump is monitored and alarmed for leakage from the tanks.

Experience with the two waste tanks in operation at West Valley has been good; there has been no evidence of leakage.

It is not planned to use the existing tanks for wastes produced by future operations at West Valley. An application by NFS[5] for modifications and increased capacity at the West Valley plant is now being reviewed. On February 24, 1976, NFS advised NRC that they plan to eliminate the necessity for HLW storage by solidifying the waste immediately after generation.[6] This would be made possible by storage of irradiated fuel for an average of 24 months before reprocessing. NFS is to provide details of the waste solidification facility.*

GE MFRP

No HLW storage tanks were included in the MFRP design. It was planned to calcine HLW immediately after production and store waste containers in a pool until the federal repository is available.

AGNS BNFP Tanks

The BNFP provides for temporary storage of liquid HLW as an acid solution in a stainless-steel tank 54 feet in diameter by 20 feet high. Net capacity of the tank is 300,000 gallons, with a 10% freeboard allowance above this level. The tank is equipped with cooling coils, ballast tanks, and air lift circulators for waste circulation and for purging of radiolytic hydrogen. The tank is also equipped with transfer jets and instrumentation for temperature and liquid-level measurements.

The tank is contained in a concrete vault lined with stainless steel. A sump with instrumentation to determine leakage is part of the liner system and provision is also made in the concrete structure for collection of possible leakage through the vault liner.

*On September 22, 1976, NFS announced that it is terminating reprocessing services.

The cooling coils are designed to remove 3.7×10^7 Btu/hr and are connected in a primary loop from which heat is transferred to a secondary system where it is dissipated in a cooling tower. Two backup water supplies are provided for the waste tanks. If the cooling-tower system fails, well water will be circulated through the secondary system to remove heat from the primary system. If both these systems fail, diesel-powered pumps will supply water from a nearby pond directly to the primary system cooling coils.

Purge air supplied to the air-lift circulators and ballast tanks is also required for dilution of the radiolytic hydrogen produced. Hydrogen concentration in the tank atmosphere is kept below 3% by volume.

The waste tank is vented through a dome on top of the tank equipped with internal baffles to minimize entrainment. The vented gases then pass through a series of cleansing processes before being released through the main stack.

An underground tank in a lined vault identical to the liquid HLW vault is provided for intermediate-level liquid waste, which contains about 0.0001 as much radioactivity per unit volume as HLW and is therefore not self-heating. The intermediate-level tank is of the same design as the HLW tank except that it does not include air-lift circulators, ballast tanks, or cooling coils. Agitation is provided by an air sparging unit. Provision is made for jetting or for installing a pump to remove liquid from the tank.

The BNFP tanks incorporate improvements over previous tank designs. The secondary containment is a fully lined vault that will hold more than an entire tank of waste. There are no obstructions on the tank bottom, so that solids cannot build up. Vapor and radioactive entrained particulates evaporated from the tanks' contents are contained and treated in the plant off-gas system. A primary waste cooling system is backed up by two other water-supply systems. In addition, temporary connections can be made to add pond water to a tank if all three cooling water supplies became inoperative.

Design and Impact of Typical Liquid Waste Storage

Three commercial fuel reprocessing plants have been built in the U.S. They are the Nuclear Fuel Services (NFS) plant at West Valley, New York, the General Electric Midwest Fuel Recovery Plant (MFRP) and the Barnwell Nuclear Fuel Plant (BNFP). Only one (NFS) has operated. The GE-MFRP was not able to start up because of problems encountered in cold checkout operation. These problems were not related to HLW handling but to the maintainability of the uranium fluorination line. The main sections of the BNFP are complete but the plutonium conversion and HLW solidification facilities have not yet been built. Licensing review is still underway.

In addition to these three existing plants, there are two new design considerations with respect to liquid HLW storage and handling. The NFS plant shut down voluntarily in early 1972 to make extensive modifications which would more than

double plant capacity. This revised plant design was based on a design approach similar to MFRP wherein liquid HLW is solidified immediately. The other design is for an entirely new plant. The Exxon Corp. has applied for a construction permit to build a complete fuel reprocessing center at Oak Ridge, Tennessee.[8] The Exxon plant design, which is also undergoing licensing review, does not include large storage tanks for liquid HLW but is designed for prompt solidification of these wastes as they are generated.

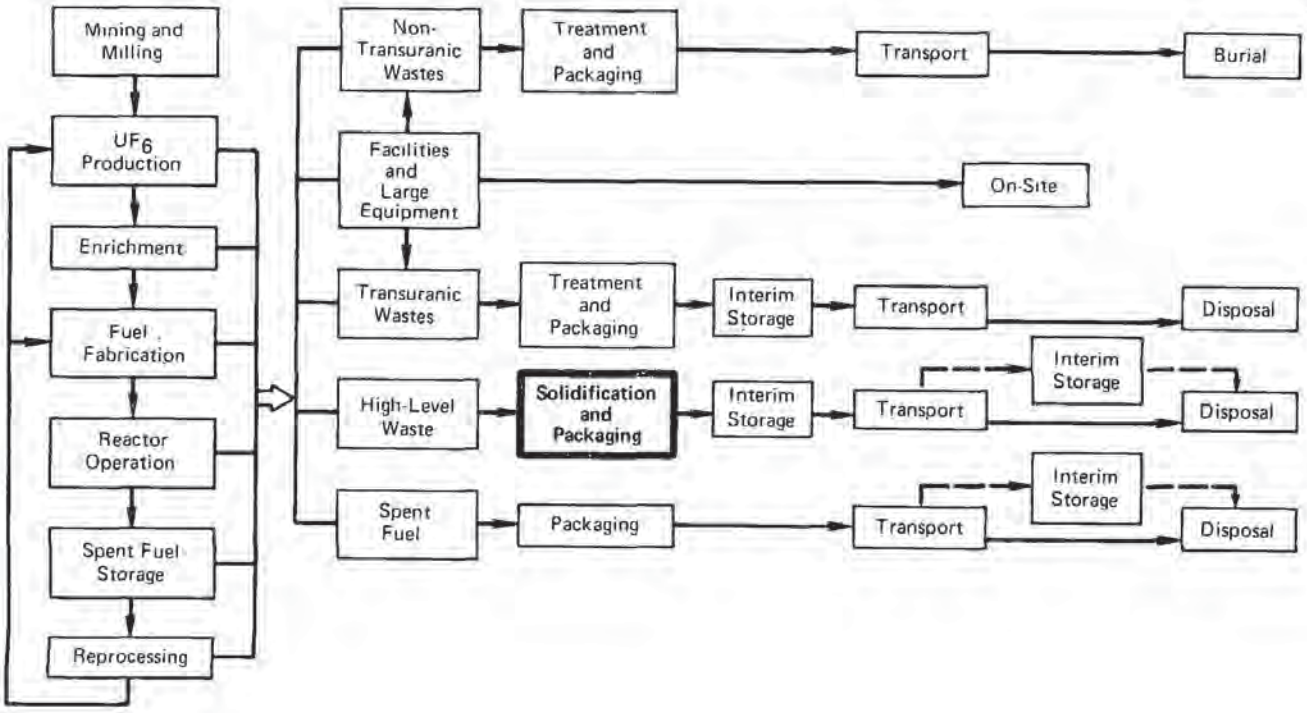
Examination of currently available commercial FRP designs shows that only the BNFP is designed for storage of large quantities of liquid HLW. The BNFP will probably be the first plant to operate if commercial reprocessing resumes in this country. The BNFP, like all such plants, is subject to the requirements to solidify liquid HLW within 5 years after generation.[9] Therefore, even the BNFP is not committed to an indefinite accumulation of liquid HLW in storage. Once the plant is operated and the waste solidification facility completed, the BNFP can be run in a manner similar to other FRP designs, with prompt solidification of liquid HLW. Thus it is concluded that large inventories of liquid HLW in storage will not be characteristic of commercial FRPs.

During normal storage, waste tanks have a gaseous effluent consisting of water, hydrogen, and oxygen produced by radiolysis, some oxides of nitrogen from the nitric acid solution, some volatile radionuclides, and entrained particulates, all typically carried in the air flow which agitates and purges the tanks. This effluent is routed to the plant gas treatment system where it mixes with and becomes masked by the gaseous effluents of the separation process (Sec. 4.1).

The probability of leakage has been reduced with the development of lined vaults for waste tanks and the use of better tank steels. Consequently, the likelihood of impact through leakage to the environment has been minimized.

Accidental dispersion of waste from the tanks has also been minimized by use of small waste storage tanks[8] which keep liquid-waste inventories low, thus reducing the potential consequences of accident. Accidental releases are further reduced by tank design. Cooling and air purge systems are carefully designed to keep waste temperatures low and to prevent the accumulation of ignitable concentrations of hydrogen. Spare tanks are provided, along with reliable transfer systems, so that waste can be transferred to a new tank if a defect is discovered.

In summary, then, the environmental impacts of liquid HLW storage at commercial FRPs are expected to be low. Stored inventories will be small and tanks are carefully designed, minimizing the risks of leakage or accidential dispersion to the environment. Routine emissions of gases and vapors are included in the emissions evaluated for the reprocessing plant itself.



4.2.2 Solidification

Present regulations (10 CFR 50, App. F) require that HLW be solidified and shipped to a federal repository. Solidification is desirable to provide protection against dispersal in untoward events. Thermal and chemical stability, insolubility, and capability to withstand impact are the key advantages to be gained by such a process.

4.2.2.1 Technology Selection

The two principal HLW solidification forms are calcine and borosilicate glass.[1] Glass has been selected for this analysis because it will produce a larger waste volume, and includes impacts from both calcination and melting.

The potential for producing a durable glass waste form was recognized over 20 years ago.[2,3] Glass development has continued, and formulations with leachability equivalent to durable minerals such as quartzite and granite have been demonstrated.[4] Tests have been conducted to verify that the glass will maintain desirable properties after devitrification and considerable radiation damage. Results indicate that the maximum effects (less than a factor of 10 increase in leach rate) occur from storage temperatures near 750°C[5] and that radiation effects for at least 2000 years do not create any significant problem.[6,7]

Equipment for glassmaking has been developed and demonstrated in hot cells both in this country[8] and abroad.[9] Vitrification generally involves a two-step process: calcination followed by melting. The principal calcining methods are fluid-bed, spray, and rotary kiln.[1] The main melting options include in-can melter, metallic melter, and ceramic melter.[1] Major emphasis in this country is being placed on the

spray calciner/in-can melter system for near-term applications.[10] The spray calciner has the advantages that there is no holdup of material within the equipment, it can be started and stopped easily, it produces a fine powder which reacts easily with glass frits, and it can handle a wide variety of feed materials, including those with high sodium content, without process upset.[11] Some of the advantages of in-can melting which led to its preference are simplicity of processing vessels and heaters, no internal moving parts, no transfer or valving of molten glass required, elimination of melter deterioration and disposal problems, no separation of materials within a melter, and operability under nonoxidized melting conditions.[12]

4.2.2.2 Description

The spray calciner/in-can melter solidification system is shown schematically in Figure 4.3. In the spray calcination process, HLW is pumped to an atomizing nozzle located at the top center of the heated-wall spray calciner. The waste is atomized by pressurized air, producing small droplets which are dried and calcined in flight through the 700°C wall-temperature vertical spray chamber. Calcination time is extremely short, thus inhibiting radionuclide volatilization and allowing only low calcine holdup in the calciner. Periodic operation of wall-mounted pneumatic vibrators prevents scale accumulation. Atomizing air, steam, and oxides of nitrogen leave the calciner through sintered stainless-steel filters which are periodically cleaned by a reverse pulse of air. Less than 0.1% of the nonvolatile radionuclides pass through the filters, thus significantly reducing the load on the effluent treatment system and reducing the amount of liquid recycled to the calciner from the treatment process.

Heated-wall spray calciner operation is relatively simple. Startup is accomplished as rapidly as the calciner furnace can be heated to temperature, which is typically 3-8 hrs. Shutdown requires only discontinuation of liquid feed, followed by a 1-minute nozzle flush. Since spray calcination does not rely on physical contact with a hot surface, waste concentration is not a process concern.

The HLW calcine generated by the spray calciner drops directly into a storage canister located in a multizone furnace below the calciner. Since calcining is a continuous operation, while in-can melting is a batch process, two melting furnaces are located below the calciner, and a diverter valve is placed in the duct connecting the calciner to the melters. The glass-forming frit is added to the calcine above the diverter. While the batch is being transferred to one canister, a filled canister is replaced by an empty one in the other furnace position.

As the melt begins to fill the canister, the temperature below the melt level rises above processing temperature because of heat generated by the waste. Therefore, appropriate furnace zones are turned off after the melt level rises above them, and air cooling is initiated in those zones to maintain processing temperatures. After a canister is filled to the desired level, it is maintained at temperature for several hours to assure complete melting. Then it is cooled, capped, and removed from the furnace. The canister is seal-welded, decontaminated, and integrity-checked before removal to storage.

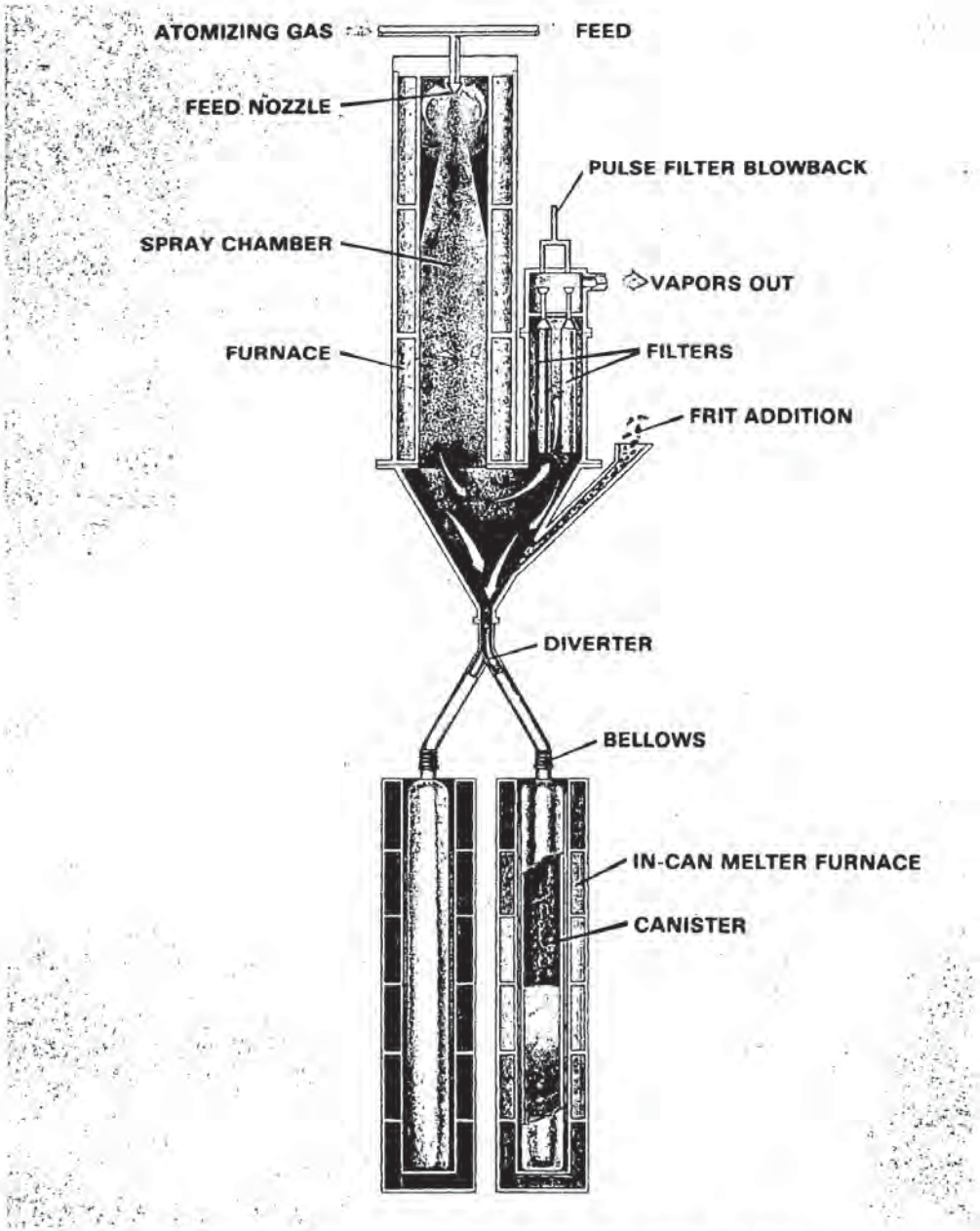


FIGURE 4.3 - SPRAY CALCINER/IN-CAN MELTER

Off-gases from the spray calciner and in-can melter are treated to remove particulate matter, volatile and semivolatile elements and compounds, and nitrogen oxides. The off-gas system comprises different process units, including condensers, scrubbers, fractionators, filters, and concentrators.

Solidification equipment is remotely operated within shielded cells in a solidification facility. These facilities are described in conceptual designs, preliminary safety analysis reports,[13] and final safety analysis reports.[14]

4.2.2.3 Bases for Impacts

Waste solidification facilities will be sized to correspond to the associated reprocessing plant. The Barnwell site is taken as a reference environment.[15] No liquid effluents are produced in the solidification facility. Gaseous effluents will be treated within the facility, then sent to the building ventilation exhaust treatment system.

4.2.2.4 Environmental Impacts

Impacts identified in the following section have also been included in the totals for reprocessing (Sec. 4.1), but are detailed here to identify specific impacts of the solidification process.

The principal nonradioactive effluent from the solidification activity will be 36 MT/yr of NO_x not removed in the off-gas system. Operation of the solidification facility will require about 800 kW of continuous power and 3,200 gpm of cooling water for product process, and will discharge 160 gpm to the atmosphere, principally from the cooling tower. Total heat rejection to the environment would be less than 60 million Btu/hr. The radiological source terms in Table 4.3 were taken from Ref. 16 and multiplied by 1.33 to account for the 2,000 MT/yr model plant size.

Since glass is resistant to shock damage and the quantities of particulates generated in this fashion are small,[17] consequences of accidents in a solidification facility involving solidified waste are minimal. Glass waste forms are also resistant to water leaching[9] and hence dispersal of material by that mode is minimized.

Accidents involving the liquid or unvitrified materials should be more severe. The leak of HLW liquid concentrate from a feed line into the cell has been analyzed[18] and found to result in a 0.013-mrem maximum dose to the individual. An explosion in a fluid-bed calciner, which has a much larger inventory of waste than a spray calciner, has also been analyzed[19] and found to result in a maximum dose of 24 mrem to an individual.

Dose calculations made for normal operation of the Barnwell facility[20] have been increased by the 1.33 factor to account for the increased capacity of the reference facility. The primary food pathway to man involves the consumption of unwashed leafy vegetables and of meat and milk from animals that ingest radioactive materials deposited on grazing areas. Maximum dose rates to the individual at the

site boundary for the reference solidification facility are shown in Table 4.4. The total annual dose to the population within 50 miles from the waste solidification facility is calculated to be 10 person-rem, compared to 64,000 person-rem due to natural background.

Table 4.3

ESTIMATED ANNUAL RELEASE OF RADIONUCLIDES
FROM NORMAL OPERATION OF THE WASTE SOLIDIFICATION FACILITY*

Radionuclide	Annual Release** Rate (Ci/yr)
H-3	4.51 E+4
Sr-89	3.23 E-3
Sr-90	1.32 E-2
Y-90	1.32 E-2
Y-91	6.45 E-3
Z-95	1.28 E-2
Nb-95	2.65 E-2
Ru-103	3.68 E-2
Ru-106	1.14 E-0
Ag-110m	3.05 E-4
Sb-125	1.31 E-3
Te-127m	9.89 E-4
Te-129	5.40 E-5
I-129	3.35 E-2
I-131	6.37 E-4
Cs-134	7.21 E-2
Cs-137	4.23 E-2
Ce-144	9.53 E-2
Pm-147	1.63 E-2
Eu-154	1.18 E-3
Eu-155	9.74 E-4
U-234	1.32 E-9
U-235	3.05 E-11
U-236	4.92 E-10

*For entire plant.
**The E format is used rather than scientific notation; i.e., 4.51 E+4 = 4.51×10^4

Table 4.3 (cont'd)

Radionuclide	Annual Release** Rate (Ci/yr)
U-238	6.12 E-10
Am-241	3.37 E-5
Am-243	3.27 E-6
Pu-238	9.18 E-6
Pu-239	6.28 E-7
Pu-240	9.28 E-7
Pu-241	1.92 E-4
Pu-242	2.55 E-9
Cm-242	1.93 E-3
Cm-244	4.12 E-4

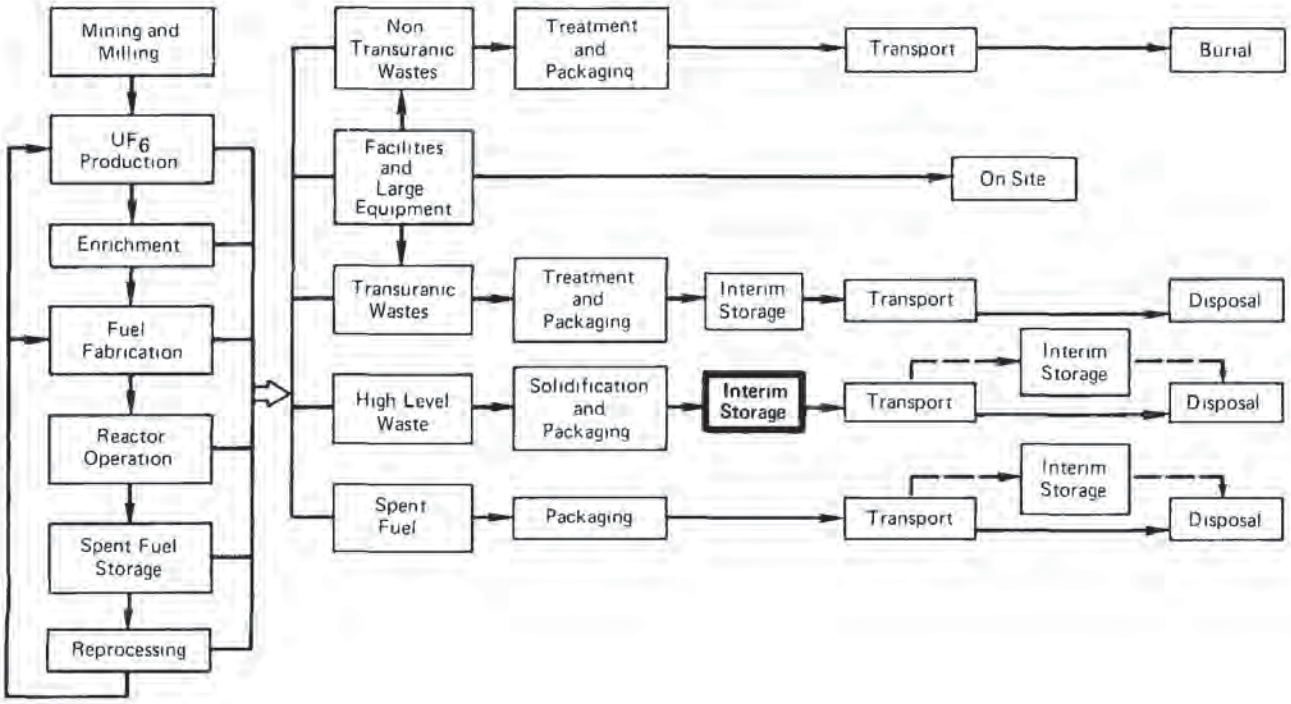
*For entire plant.
**The E format is used rather than scientific notation; i.e., 4.51 E+4 = 4.51×10^4

Table 4.4

MAXIMUM ANNUAL DOSE TO INDIVIDUAL AT THE SITE BOUNDARY FROM NORMAL AIRBORNE EFFLUENT FROM WASTE SOLIDIFICATION FACILITY*

Adult Organ	Dose (Millirem)
G. I. tract	1.7
Bone	.7
Thyroid	1.1
Lung	.5
Total body dose	.52

*For entire facility operating at capacity.



4.2.3 Interim Storage of Solidified HLW at a Reprocessing Plant

4.2.3.1 Technology Selection

The interim-storage water basin facility at the FRP will receive and retrievably store solidified high-level radioactive waste produced in the FRP using operating methods based on established and proven technology.[1,2,3]

It is expected that liquid HLW will be converted to borosilicate glass (Sec. 4.2.2) in a solidification facility at the FRP, and that the solid waste will be stored at the FRP for up to ten years, and will then be shipped to a federal repository. During interim storage the decay heat must be removed continuously to keep the canisters at a safe storage temperature (centerline temperature less than 800°C[4]). Several cooling concepts for interim storage of solidified HLW canisters, such as water cooling and air cooling, are discussed in Ref. 1.

For this Supplement, the reference method is storage of solidified HLW canisters in racks in a water basin. This method is chosen because it permits storage of waste with high heat content per unit volume, and because the nuclear industry has had about 30 years of experience with the use of water basins for storage of radioactive materials.

Water storage basins are designed to protect personnel from radiation by using water as a shield; transfer decay heat to the water and use water-to-water heat exchangers to keep the basin at prescribed temperatures; serve as a confinement barrier for solid waste that may escape from the canister; and allow direct viewing of waste canisters.

4.2.3.2 Description

Canisters are stored in water-filled, stainless-steel-lined basins. Radioactive decay heat is transferred from the waste to the basin water and then released to the atmosphere by cooling towers as shown in Figure 4.4. Basin water is maintained at about 120°F under normal operating conditions. Pumps, heat exchangers, and associated systems are designed to provide redundancy and for ready replacement since the water could reach boiling temperature if the cooling system became inoperative for a day or more. To prevent even the slow loss of water from the basin due to boiling, emergency water systems such as a standby emergency well and pump are provided as additional backup features.[1-3,5-7]

To minimize corrosion of the stainless-steel canisters and the basin liner during storage, water purity is maintained by passing part of the recirculated cooling water through a filtration and demineralizer system. A high-capacity water cleanup system may be provided for use if the water becomes contaminated.[1-3,5-7]

4.2.3.3 Bases for Impacts

A preliminary safety evaluation of the water basin concept was made to evaluate engineered features which prevent or mitigate the consequences of potential safety hazards.[8] The evaluation is based on calcine as the waste form since the conclusions drawn regarding waste releases are conservative, as a glass form is less leachable and is less dispersible than calcine by a factor of 100 to 100,000.[9,10]

Radioactive releases during normal operation result from minor routine equipment or process failures. Abnormal occurrences result from equipment failures, operator errors, or unplanned process variations that may occur during plant operations. Occurrences analyzed include canister failure in a storage basin, heat-removal equipment failure, and leakage of primary coolant.

An assumption governing plant design is that during normal water basin operations, radiation exposures to the public, and radionuclide concentrations in plant liquid and gaseous effluents, will be no greater than 10 percent of the maximum limits prescribed in 10 CFR 20.[11]

Accidents that have been considered include (1) canister failure in a dry cell, (2) fire or explosion, and (3) loss of cooling capability. In addition, analyses were performed for externally induced events, i.e., earthquakes and tornadoes. Design features which reduce the probability of occurrence and possible magnitude of releases of radioactive material resulting from these events are: (1) additional HEPA filtration capability, (2) purging and containment of hydrogen generated in the basin, (3) high-capacity ion-exchange systems for grossly contaminated water basins, (4) structures and systems designed to withstand maximum (design basis) tornadoes and earthquakes, and (5) redundant systems included as backup measures; e.g., cooling systems.

The maximum accident analyzed for interim storage is a canister drop in the dry transfer cell that separates the waste solidification facility from the water basin. The analysis shows that if such an accident occurs, plant effluent radionuclide

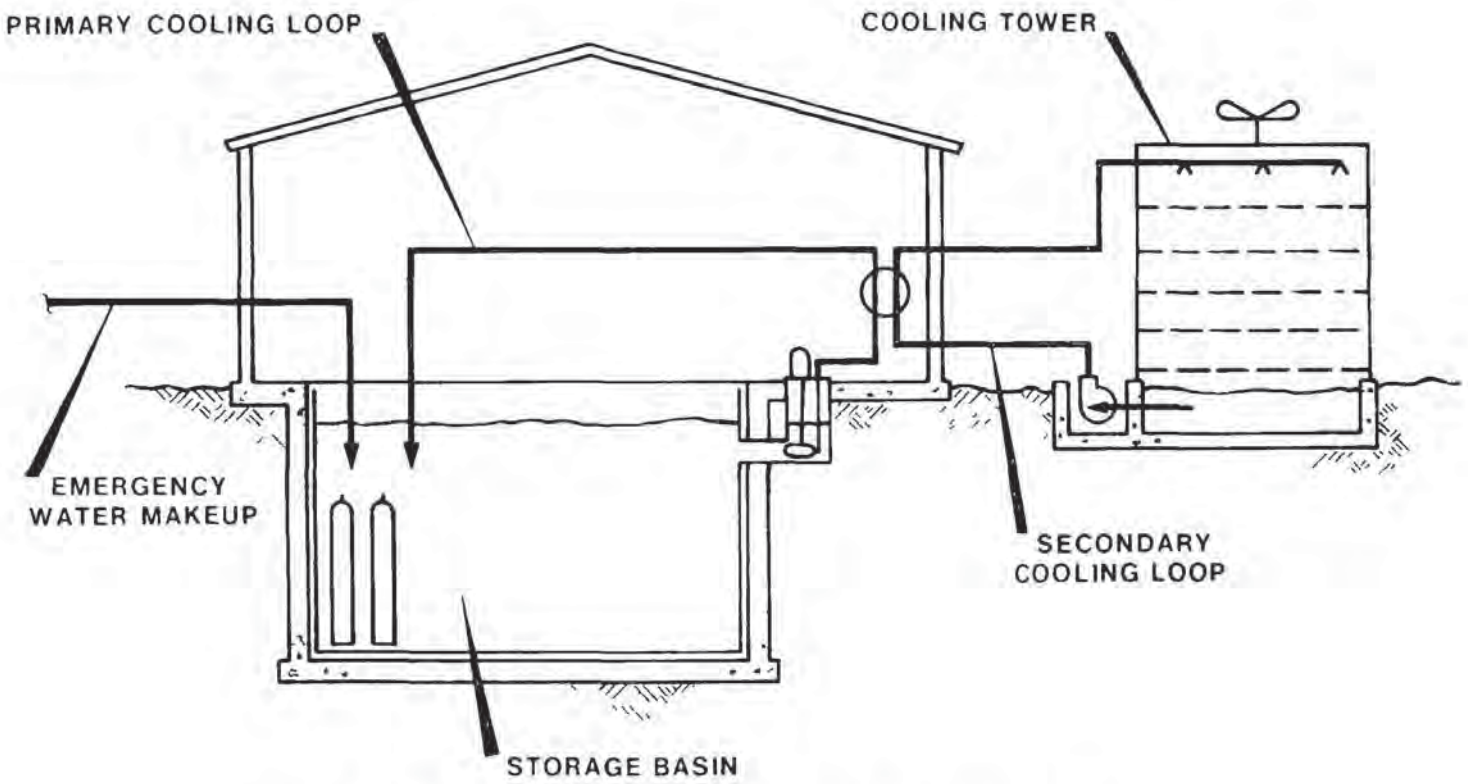


FIGURE 4.4 - HEAT REMOVAL SYSTEM; WATER BASIN CONCEPT

concentrations and/or radiation exposure to the general public could exceed 10 CFR 20 limits. Release from a dropped canister is discussed in Section 4.2.5.3.

4.2.3.4 Environmental Impacts

The environmental impacts of interim storage are included in the total impacts of the FRP, which are discussed in Section 4.1. Impacts from the interim storage of solidified HLW in water basins are presented in this section to identify the portions of the FRP total environmental impact that are attributable to interim storage.

The interim storage facility would be located adjacent to the FRP, and would require about 4 acres of land.[12]

The water cooling system would be a recirculating system that would require water additions to allow for evaporation in the cooling towers. The total estimated amount of water needed for interim storage is 150 million gallons per year (285 gpm).[13]

Approximately 100 megawatts per year of radionuclide decay heat will be dissipated to the atmosphere from interim storage facilities via cooling towers.[14]

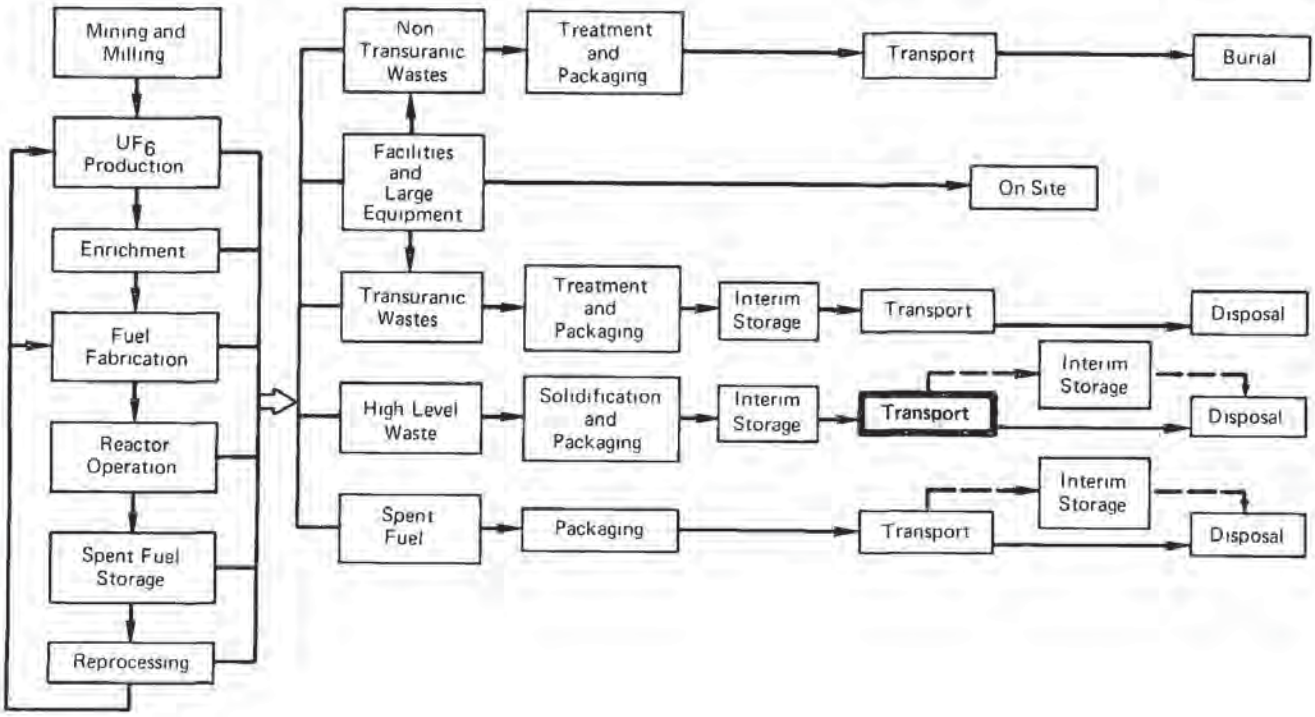
Under normal operating conditions, radionuclide releases to the atmosphere will result in air concentrations well below acceptable concentrations.[11] Occasional releases to the exhaust air system might come from cleanup operations around storage basins. The storage area would normally be a clean, uncontaminated working area and the ventilation exhaust air from it would contain negligible amounts of radioactivity.[1]

An analysis performed for water basin storage of canisters at an RSSF concludes that abnormal occurrences will not result in exposing operating personnel or the public to annual average effluent radionuclide concentrations in excess of the limits prescribed in 10 CFR 20.[11]

No radiological releases to waterways or groundwater will occur, as all contaminated water is recycled to the FRP for processing in a waste-treatment system. Filters and ion-exchange columns will be routinely used to maintain the cleanliness of circulated basin water[2] and will be disposed of at an approved disposal site.

Gaseous nonradioactive waste would be principally combustion products from the oil-fired power plant which, it has been estimated, would consume the equivalent of about 100,000 gallons of low-sulfur fuel oil per year attributable to interim storage. Combustion products released to the atmosphere would be a maximum of about 0.8 ton per year each of oxides of nitrogen, sulfur dioxide, hydrocarbons, and particulates.[15]

Ordinary industrial trash and miscellaneous nonradioactive solid wastes will be compacted and transported to landfill burial. The principal impact would be the use of about 0.1 acre per year of land for burial.[16]



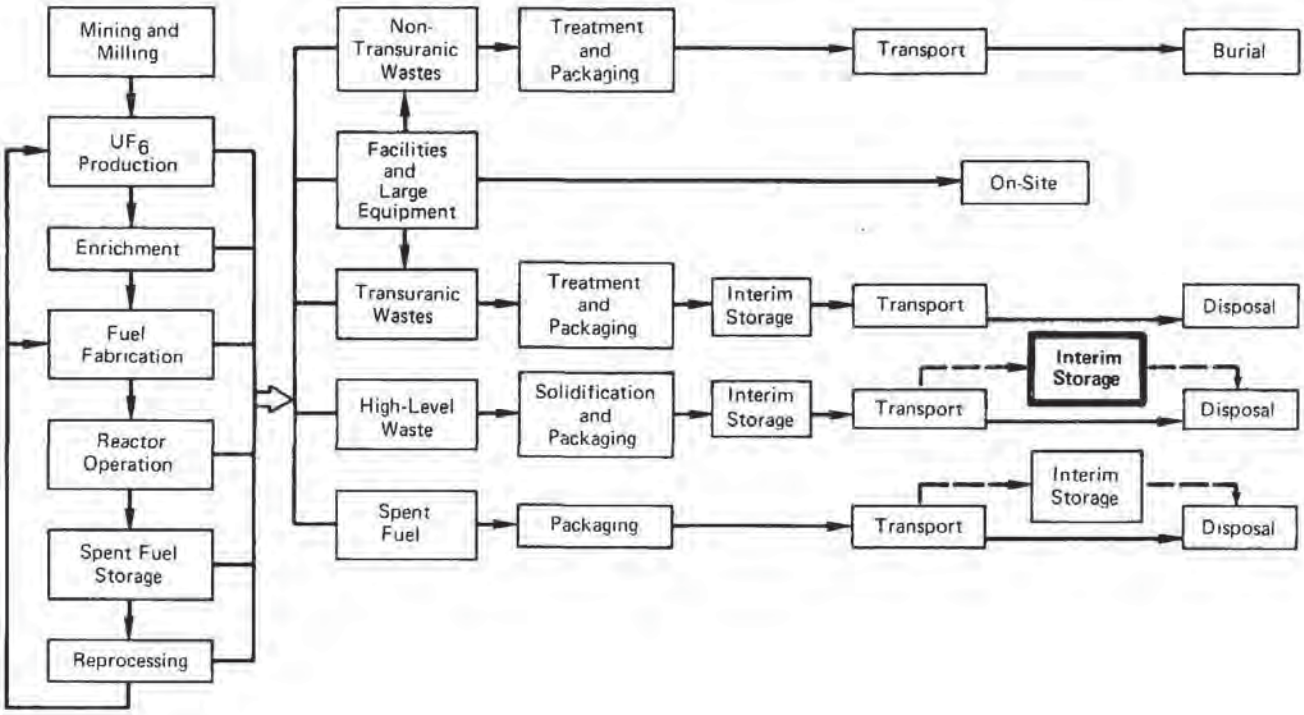
4.2.4 Transportation

Environmental effects resulting from transportation of radioactive wastes are discussed in Section 4.9.

Two transportation steps are required for HLW: (1) between the FRP and the interim storage facility (RSSF); and (2) between the interim storage facility and the federal repository. The characteristics of each step (miles per shipment, waste volumes, number of shipments, etc.) for both the uranium cases are given in Table 4.33.

Information presented in Section 4.9.4 indicates that the thermal and radiological effects of routine HLW transport are negligible.

Transportation casks for solidified HLW are assumed to have structural and containment features similar to those of casks for transporting spent fuel.



4.2.5 Interim Storage at a Retrievable Surface Storage Facility (RSSF)

Introduction

Final geologic disposal sites for solidified HLW are currently under evaluation. Studies have also been conducted within ERDA to provide a central federal repository (a retrievable surface storage facility--the RSSF) for interim isolation of waste until geologic disposal becomes available. This section discusses the RSSF for the storage of solidified HLW since it may be a contingency measure in HLW management.

Principal objectives for an RSSF are:

- To provide safe HLW storage for 100 years.[1]
- To assure public health and safety from HLW radiation.
- To assure safe facility operation and waste retrievability under postulated accident conditions (externally and internally induced).
- To assure protection of the environment.

Principal design bases to ensure that the RSSF would achieve the objectives are (1) storage integrity and confinement of waste; (2) radiation protection; (3) heat removal; and (4) waste retrievability. The RSSF would be located on an existing ERDA site.[2]

4.2.5.1 Technology Selection

Three basic engineering concepts for an RSSF are presented in Refs. 2, 3, and 4. For all of these concepts, the waste receiving and service facilities of the RSSF are similar. The concepts differ in the cooling methods used to remove decay heat from the waste.

- Sealed Storage Cask Concept[5]

  All canisters are placed in thick-wall, high-integrity overpacks (casks) which provide waste confinement, gamma shielding, and a passive cooling system. Neutron and additional gamma shielding for each unit is provided by thick concrete outer shields. Heat is removed by air which flows naturally through the annulus between the cask and the concrete shield.

- Water Basin Concept[6]

  Bare waste canisters are stored in water basins and are dependent on the water, the building structure, and mechanical cooling systems to provide shielding, physical protection, and cooling.

- Air-Cooled Vault Concept[7]

  All canisters are placed in thin-wall, high-integrity overpacks and stored in shielded vaults designed to induce natural-draft air cooling.

The sealed storage cask concept (SSCC) was recommended for implementation as the RSSF by a special panel,[4] and is therefore selected as the reference RSSF for this Supplement. The SSCC uses natural air flow to cool the waste, avoiding reliance on water supplies, powered coolant-circulation systems, and massive fixed concrete storage structures, thus simplifying both site operation and eventual decommissioning. The concept is described in Refs. 5, 8, and 9 (Fig. 4.5).

4.2.5.2 Description

The waste is received in stainless steel canisters sealed inside carbon-steel storage casks. The storage cask is placed inside a concrete shield with a 6-inch-wide annulus between the storage cask and the inner face of the concrete shield to permit convective air flow to remove decay heat from the waste. The assemblies (called storage units) are placed on concrete pads in an outdoor storage area, spaced at about one per 400 ft^2.

Facilities

The RSSF will have two principal facilities: the receiving and assembly building, and the storage area.

HLW will be received at the RSSF in shipping casks. The receiving and assembly facility will include the following areas: (1) receiving area,

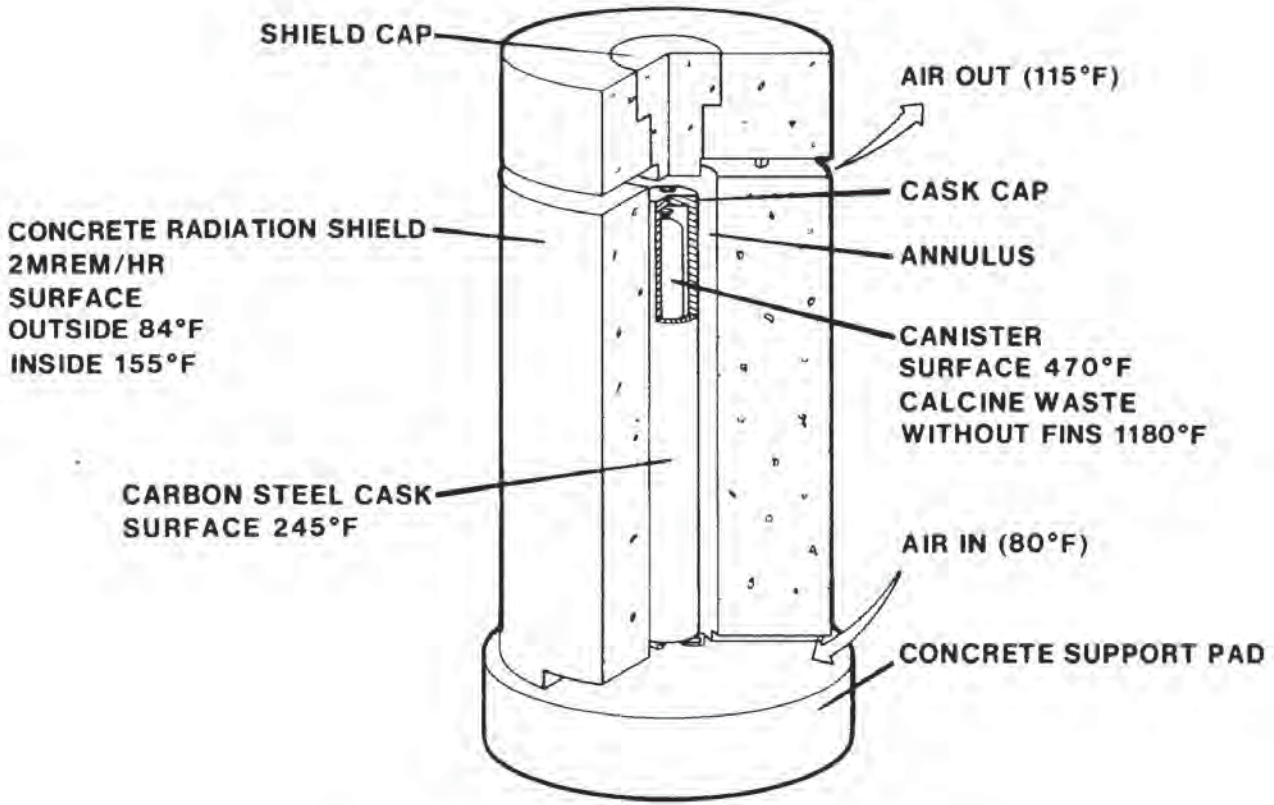


FIGURE 4.5 - SEALED STORAGE CASK CONCEPT (SSCC)

(2) cask weld and test area, (3) storage unit assembly area, (4) transfer corridor, (5) support area, and (6) personnel areas. Waste collection and processing systems, building services, and control areas are located in the receiving and assembly building to handle requirements for both the receiving and storage functions.

Waste canisters are unloaded from the shipping casks remotely in the receiving cell, examined, and, if necessary, decontaminated in the spray chamber. Defective canisters are placed into a carbon-steel overpack and equipped with a temporary seal before transfer to the weld and test cell. The receiving cell includes a compartment where the waste canister is inserted into the transfer cask.

Ventilation systems for radioactive areas are controlled so that air flows from occupied regions toward operating areas of increasing radioactivity and is exhausted through two sets of HEPA filters in series.

The cask weld and test cell is a gamma-neutron shielded facility two stories high enclosing approximately 1,500 sq ft. Features include equipment to remove the canister from the RSSF transfer cask, introduce carbon-steel storage casks into the cell, place the canister in the storage cask, and weld, stress-relieve, and test the cask.

The storage unit assembly area provides the fixtures and handling equipment required to (1) receive subassemblies of the storage unit, (2) assemble and inspect it, (3) transport it into and out of the storage unit cell, and (4) place it on a tractor-trailer for transfer to the storage area. After transfer, the unit is placed on a storage pad by a crane. The pads are arranged on a nominal 17 ft by 24 ft module to permit retrieval of any storage unit at any time (Fig. 4.6).

The storage area is subdivided into lots accommodating 1,000 storage units. This allows a spacing of approximately 400 sq ft per unit and limits the radiation dose to design conditions.

4.2.5.3 Bases for Impacts

Reference RSSF Site

Since site location is undetermined, evaluation of environmental impacts is based on a western site located on existing ERDA-controlled land, as discussed in Ref. 10.

A preliminary safety evaluation of the RSSF, based on calcined HLW, indicated that under normal and abnormal conditions, radionuclide releases to the environment will be well below the maximum allowable for compliance with applicable NRC radiation dose and effluent concentration guide limits.[4,11,12]

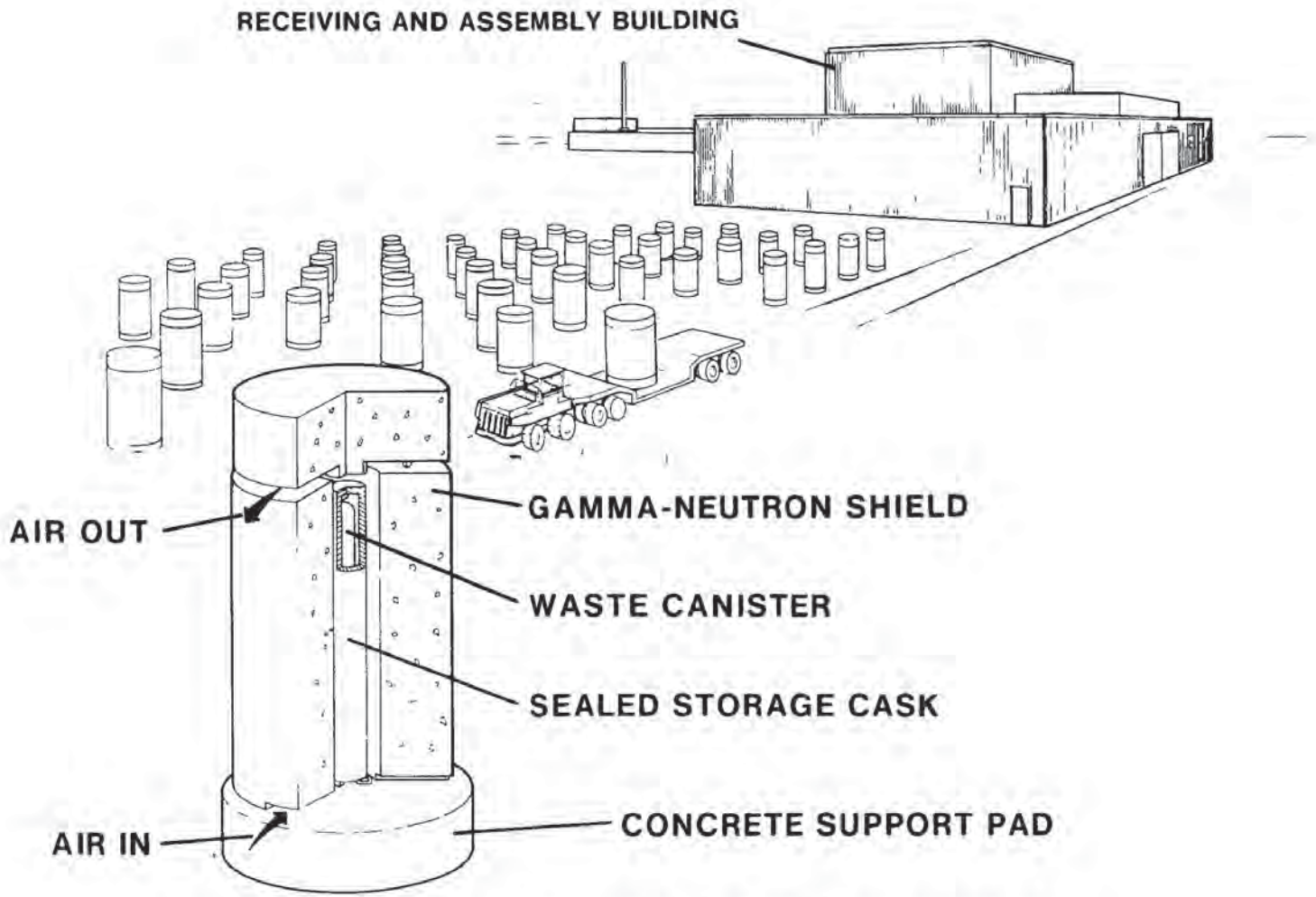


FIGURE 4.6 - RETRIEVABLE SURFACE STORAGE FACILITY (RSSF)

In this Supplement, a borosilicate glass waste form is used because it is much less desirable and leachable than the calcine form.[13,14] Since it is a less-dense waste form, it would require more canisters for packaging and eventually more land for storage. Therefore the RSSF impacts analyzed in this Supplement, based on calcine for disposal and glass for land use, are conservative.

Routine radioactive releases via ventilation exhaust air will result from air contamination due to receipt of contaminated shipping casks or canisters, decontamination operations, and other plant activities. Atmospheric effluents will contain no more than 0.005 curie per year measured at the plant stack.[15] It is estimated that the maximum routine concentration of TRU radionuclides leaving the stack would be less than 1.0×10^{-13} microcurie per milliliter and that of the fission products would be no greater than 1.4×10^{-11} microcurie per milliliter.[15] Both of these concentrations are less than 2 percent of the occupational concentration guide limits listed in 10 CFR 20. Dispersion and dilution would normally result in concentrations below the nonoccupational limits at the site boundary.

Contaminated solid waste from decontamination work or other contact with contaminated surfaces will amount to about 50 cu ft per day,[16] and will consist of disposable clothing, rags, paper products, failed equipment, and spent filters and ion-exchange resins. These materials will be packaged and shipped to an approved low-level disposal site (Sec. 4.7).

The safety evaluation in Ref. 2 includes analyses of abnormal operating conditions such as receipt of contaminated or failed canisters, canister failure in storage, and failure of heat-removal equipment. The effects of such events upon the operation and capability of the facilities to continue to provide confinement of radioactivity during and after such events are discussed in Ref. 17. Such provisions include decontamination and overpacking of contaminated canisters in the receiving cell, sealing the canister inside the storage cask to provide an additional confinement barrier, and use of a passive cooling system to obviate the need for heat-removal equipment. The discussion concludes that abnormal operating conditions will not result in exposures of operating personnel or the general public to radiation in excess of the limits presented in 10 CFR 20.

The safety evaluation in Ref. 2 includes analyses of low-probability events such as tornadoes, earthquakes, etc. A discussion of storage-unit resistance to such events is discussed in Ref. 18.

4.2.5.4 Environmental Impacts

Accident Analysis

The postulated most severe accident that could occur at the RSSF results from a canister failure in the receiving cell. Canister rupture is postulated to occur because of accidental dropping or other shock, although receiving cell and canister design will minimize the likelihood and severity of such a rupture. It is assumed that the entire contents of calcine in the ruptured canister are released into the cell as

granular particles. The radioactive material escape path to the atmosphere would be via three sets of HEPA filters in series.

The worst-case, first-year bone dose to any individual 1,000 meters from the facility stack during the hypothetical maximum accident (about 10 millicuries released to atmosphere) has been estimated to be about 85 mrem.[19] For comparison, the corresponding whole-body dose is well below that received annually by the general public from natural background radiation exposure, which varies in the United States from about 100 mrem to about 250 mrem.[20]

Land Use

Land occupied by the RSSF facilities will be within a fenced exclusion zone. A buffer zone around the exclusion area will be needed to restrict public access. To provide RSSF storage for the waste from 507 model LWRs at year 2005 on an ERDA site, total land requirements are 650 acres.[21] Location of the RSSF on existing ERDA sites in the West is discussed in Ref. 22. Since the three alternative sites described have large tracts of unused land that would be suitable for the RSSF, added impact on land use would be minimal.

Water Use

Total water requirements for the RSSF have been calculated to be 27 million gallons per year at year 2005, based on storage waste from 507 model LWRs.[23] Ref. 24 concludes that all the principal ERDA sites have water supply sources more than adequate to meet SSCC needs without adverse effects on present or predicted long-term regional population water requirements.

Thermal Effects

Heat produced by decay of nuclides in the waste would be rejected directly to the atmosphere by natural convective cooling. The amount of heat released to the environment from the RSSF during the year 2005 has been calculated to be 72 MW/yr, based on storage of the waste from 507 model LWRs.[23] The heat-generation rate would be reduced by decay to about 11 MW at year 2100.[23]

Impact of heat released from an RSSF located on an existing ERDA site in the West is discussed in Ref. 25. Pertinent parts of this discussion are herewith summarized:

A "heat island" effect would be discernible, with a small temperature differential of 1-2°C between the storage area and the surrounding site, but the effects would probably be limited to thermal plume rise, possible "dust devil" vorticity, and formation of atmospheric steam during precipitation in the winter months. No modification of local weather would be expected. The amount of heat released per unit area would be approximately the same as the man-made heat which characterizes today's urban areas. The discussion concludes that these heat releases would have no detrimental effects on the public or the environment, and negligible effects on or near the RSSF site.

Radiological

Normal RSSF operation would result in the continuous release of very small amounts of radionuclides to the environment. Estimated population doses within a 50-mile radius would be less than 5×10^{-3} person-rem per year for the projected population in the year 2005. (This may be compared with a population dose of greater than 35,000 person-rem per year from natural background.)[26] The radiation field in the storage area would not exceed 10 mrem per hour.[27] Minor radiation damage to animal life could result from the outdoor storage, inasmuch as insects and (perhaps) very small rodents and reptiles could get into the annuli between casks and shields where gamma radiation fields would be as high as 2.5×10^{4} rem per hour.[28] However, preventive measures such as screening would minimize intrusions, and there would be no discernible effect on the overall populations of any fauna.

During normal operation of the facility, small amounts of radioactive materials will be released to the air from process building vents and from neutron activation in the annuli of storage units.[29,30] All radioactive liquid waste materials will be processed, solidified, and stored as low-level waste (Sec. 4.7). Therefore, the only exposure pathways for biota other than man will be air submersion, inhalation, consumption of depositions on forage, and direct radiation. Radiological impacts on biota other than man resulting from operation of an RSSF located at a western site are discussed in Refs. 29 and 30. The discussion concludes that there is no evidence of discernible concentrations of induced radionuclides in the food chains.

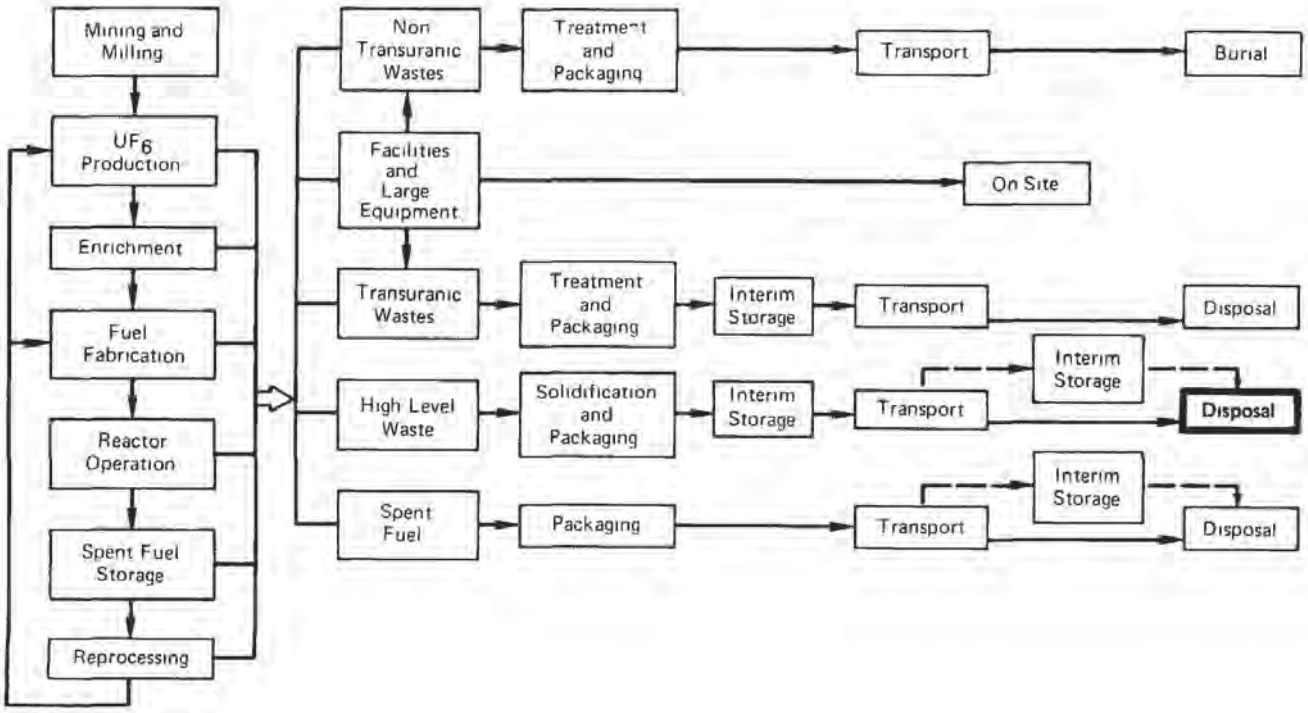
Chemicals, Liquids, and Solid Wastes

Gaseous nonradioactive waste would be principally fuel gases from the oil-fired heating plant, which it has been estimated would consume about 250,000 gallons of low-sulfur fuel oil per year. Combustion products released to the atmosphere would be a maximum of about 2 tons per year each of oxides of nitrogen, sulfur dioxide, and particulates. Such emissions are expected to have no discernible effect on the offsite environment.[31]

Liquid wastes, if radioactive, would not be released from the plant, but would be processed into a solid matrix which would then be disposed of along with other solid low-level wastes (Sec. 4.7). The total amount of low-level solid radioactive wastes to be packaged is estimated at 180 m^3 per year for the RSSF.[21] All other liquid wastes leaving the building (including blowdown from the small cooling tower serving the utility water supply) would be sent to the outdoor evaporation pond and released to the atmosphere as vapor, including waste effluent from the package sanitary waste unit. No liquid wastes would be released to the soil, but the cooling tower would reject heat and water vapor to the atmosphere. Impacts on the offsite environment of all liquid nonradioactive waste disposal would be negligible.

Ordinary industrial trash and miscellaneous nonradioactive solid wastes would accumulate and require disposal. The principal impact would be the use of no more than one-eighth of an acre per year[32] for burial.

Evaporation pond(s) would require periodic cleanouts. An estimated 21 tons per year of solid chemicals would be removed and buried in landfill. These would be primarily water-treatment chemicals. In addition, 15 to 18 tons of solid sludge would be produced by the package unit for sanitary waste disposal and would also be buried in the landfill site.[33]

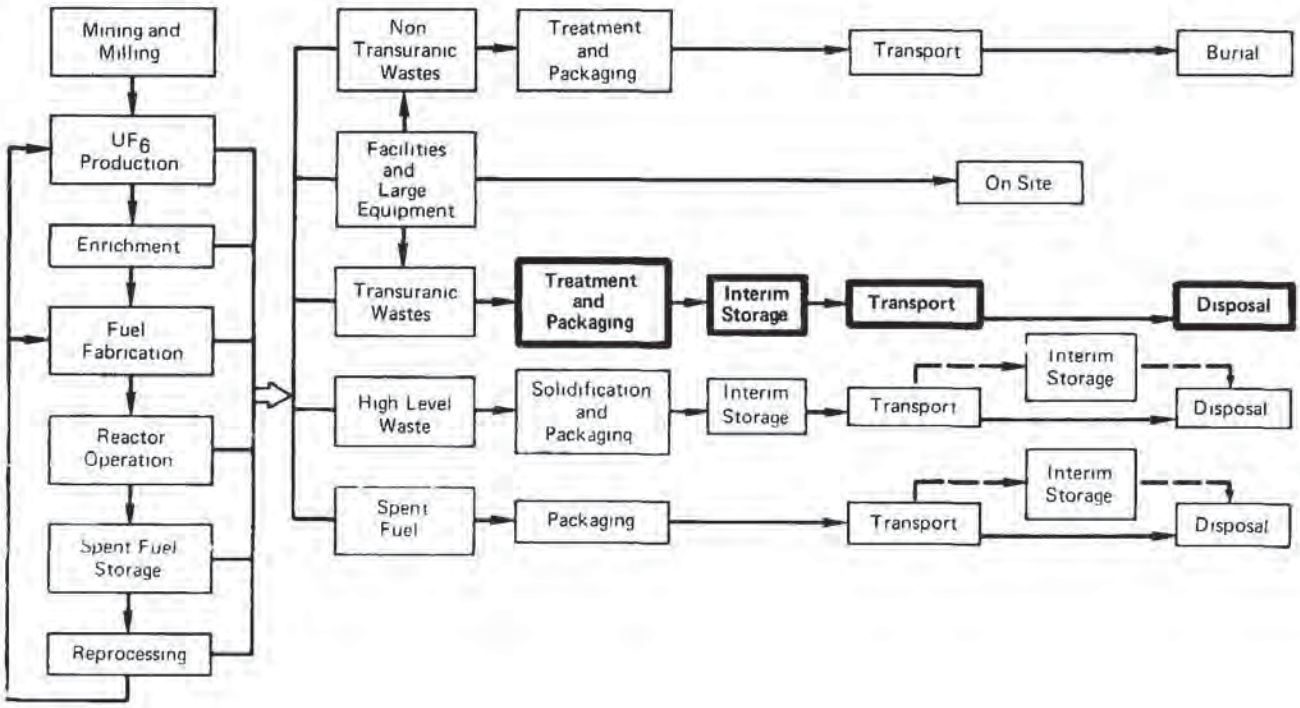


4.2.6 Disposal at a Federal Repository

Environmental effects resulting from disposal of long-lived radioactive wastes are discussed in Section 4.4.

The geologic isolation system selected for analysis is disposal with conventional mined emplacement in a bedded-salt formation.

Small amounts of radioactive materials are released to the atmosphere during normal operation of the repository. The only source for radioactive release considered significant is the gaseous effluent, which will contain a small number of radioactive particles.



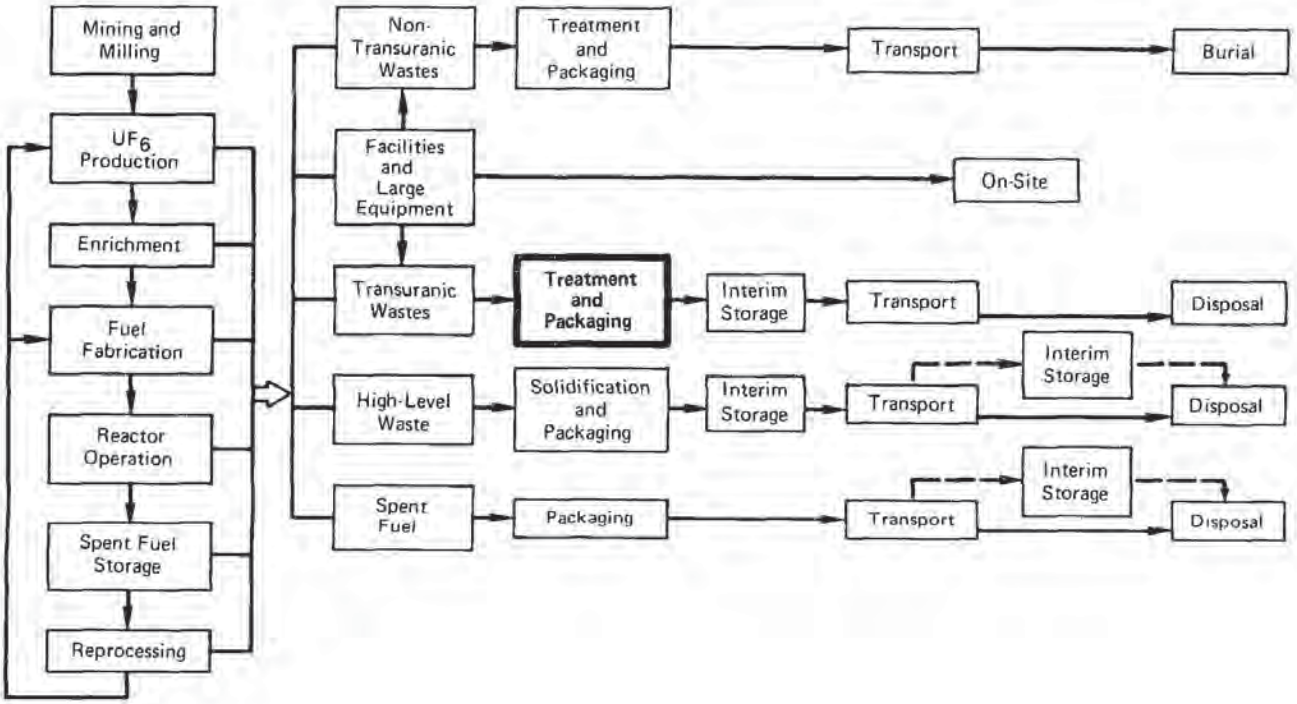
4.3 Management of TRU-Contaminated Wastes

Solid wastes contaminated with transuranics (TRU wastes) are derived primarily from the operation of fuel reprocessing plants. They consist of solidified liquids, HEPA filters, cladding hulls, and general trash. Several ways of packaging and disposal are available for these wastes, as described in ERDA 76-43, Volume 2.[1] Final management of TRU waste will be interim storage followed by consignment to a geologic repository. The waste management system, therefore, should produce waste forms compatible with final disposal conditions to avoid future reprocessing.

For the purposes of this Supplement, we expect that TRU wastes will be treated by typical methods and committed to onsite interim storage as described in docketed license applications and environmental impact statements for FRPs. TRU waste will be shipped to a federal repository for ultimate storage. Additional treatment over that proposed in the license applications would be used to stabilize and/or reduce the volumes of combustible and reactive waste fractions to increase handling safety in later storage operations.

Waste types, quantities, and reference treatment systems are summarized in Table 4.5. Waste quantities are derived from GESMO, Chapter IV, Sections D and H, supplemented by information from ERDA 76-43, Vol. 1, Section 2. The selection criteria for treatment, packaging, and interim storage, and available alternatives that were considered, are described below.

Processing and interim storage facilities required for the management of TRU-contaminated wastes are substantially less complex and smaller than the related major processing facility--the FRP. Consequently, environmental impacts of TRU waste management are relatively minor increments of those generated by the major process functions producing TRU wastes.



4.3.1 Incineration of Combustible TRU Wastes

A portion of the TRU wastes is combustible. This property presents both a potential hazard in case of fire, and an opportunity for reducing volume.

4.3.1.1 Technology Selection

The selected management system for TRU combustible wastes is based on controlled-air incineration.[1, 2, 3] This technology was chosen since combustion is a proven way of reducing waste volume and rendering the residue chemically inert.[4] Also, alternative technologies such as compaction or fixation of shredded waste in concrete would not result in maximum volume reduction or the complete elimination of combustibles.

The simplest alternative to incineration is storage in metal drums. However, rising storage and transportation costs may provide increasing incentive to reduce waste volume. Further, fire hazard control is dependent on container integrity.

Compaction is the most widely used volume-reduction technique in the nuclear industry;[1,5,6] reductions of three to five-fold are routinely obtained. However, while compaction lowers waste combustibility, it does not eliminate the possibility of fire.

Table 4.5

TRU WASTE GENERATION

A. Reference Fuel Reprocessing Plant (FRP): 2000 MTHM/Year, Serving 57 Reference Reactors

Waste Category	Treatment Process	Plutonium Content			Annual Volume; m^3/RRY		Equivalent No. of 55-Gal. Drums/RRY	
		% of Plant Throughput	U-Only Recycle, g/m^3	U + Pu* Recycle, g/m^3	Before Treatment	After Treatment	Before Treatment	After Treatment
Fuel Bundle Residues	Pack in sand matrix	0.1	18	37	15	15	75	75
Failed Equipment	Decontaminate and disassemble	0.01	8	16	3.5	3.5	26	18
Ventilation Filters	Compact into drums	0.21	270	560	2.1	0.5	30	3
General Trash Combustible	Incinerate	0.04	7	15	15	0.7	75	3
Noncombustible	Compact and pack with failed equipment	0.01	7	15	3.7	1.8	19	9
Liquids and Dispersible Solids	Package in cement matrix	0.03	17	34	4.5	7.4	23	37
Organic Solvent	Incinerate	0.001	7	15	0.4	-	2	-
TOTAL		0.4	25	51	44	29	250	145

*Included for completeness only.

Table 4.5 (cont'd)

B. Reference Mixed Oxide Fuel Fabrication Plant (MOX-FFP):* 360 MTHM/Year, Serving 34 Reference Reactors

Waste Category	Treatment Process	Plutonium Content			Annual Volume; m^3/RRY		Equivalent No. of 55-Gal. Drums/RRY	
		% of Plant Throughput	U-Only Recycle, g/m^3	U + Pu* Recycle, g/m^3	Before Treatment	After Treatment	Before Treatment	After Treatment
Failed Equipment	Decontaminate and disassemble	0.003	-	8	2.1	2.1	16	11
Ventilation Filters	Compact into drums	0.06	-	540	0.6	0.2	9	1
General Trash								
Combustible	Incinerate	0.04	-	64	3.3	0.2	16	1
Noncombustible	Compact/pack with failed equipment	0.01	-	64	0.8	0.4	4	2
Liquids and Dispersible Solids	Package in cement matrix	0.004	-	28	0.8	1.5	4	8
TOTAL		0.12	-	83	7.6	4.4	49	23

*Included for completeness only.

Encapsulation of wastes by shredding and mixing them with a noncombustible matrix material such as cement[7] eliminates combustibility and reduces the mobility of contained radioisotopes. Volume reduction is negligible, and the addition of concrete significantly increases the total mass of waste to be handled.

Combustible nuclear wastes have been incinerated in the United States and in foreign countries for over 25 years.[1,4,8,9] The various methods used have reduced waste volume by factors of 20 to 50, depending on the process, waste composition, and the as-generated waste density. Recent technological advances have effectively eliminated most earlier system deficiencies[2,8,9] such as incomplete combustion, clogged off-gas systems, and corrosion.

Controlled-air incineration with a high-energy, aqueous off-gas system was chosen because it represents "state-of-the-art" waste-management technology[3] and has been used in applications ranging from disposal of pathological wastes to combustion of scrap rubber and plastics. It has several advantages over earlier industrial incineration methods, including:

- low particulate emissions,
- operational flexibility to accept a wide range of waste types,
- ease of combustion-rate control.

Advantages of a high-energy, aqueous, off-gas system include but are not limited to:

- operational flexibility to satisfactorily attenuate a wide range of gaseous and particulate emissions,
- highest efficiency particulate cleanup system,
- widely demonstrated technology.

In brief, the controlled-air system chosen for use with TRU combustible wastes is based on proven technology, modified to meet radiological health and safety standards.

Several other promising incineration concepts are under development, including a fluidized-bed concept,[10] a cyclone incinerator,[1] acid digestion,[11] and a molten-salt system.[12] Any or all of these systems, when developed, could become acceptable alternatives to the controlled-air concept.

The inert residue or ash remaining after combustion will be assayed for TRU content to maintain accurate processing and storage records. Cement paste will then be added to the ash to prevent dispersal of contamination.

Alternate methods of ash immobilization such as bitumenization and vitrification are available or under development.[1] Fixation in bitumen or cement are the two commonly used methods for solid wastes. Because bitumen reintroduces combustibility to the fixed ash, the cement additive was chosen for discussion in this section.

4.3.1.2 Description

The controlled-air incineration system is designed to accept the wide range of waste compositions anticipated for the LWR fuel cycle. The equipment can handle up to 100 percent of any one of the solid-waste components (cellulosics, plastics, rubber). Liquid receiving and feed equipment (separate from solid-feed handling) is provided for TRU solvents.

The complete treatment process is best described by consideration of the following subsystems: feed preparation, feed introduction, incineration, off-gas treatment, ash removal, and ash immobilization and packaging.

The feed preparation subsystem includes the following functions:

- Assay[13] of the waste to permit separation into TRU and scrap* fractions
- Removal of noncombustible items[14,15]
- Preparation of large or dense waste items by cutting or shredding.[15]

The incinerator described in this section will be used solely for TRU combustible waste. Noncombustible TRU will be treated by methods described in Section 4.3.2.

To transport waste from a holdup location at the end of the feed preparation line to the incinerator, a ram feeder, similar to a horizontal piston, is used.

The incinerator is a conventional, commercially available, dual-chamber unit, modified for radioactive service. Air is used in the first (ignition) chamber to incinerate the solid wastes under substoichiometric (oxygen-deficient) conditions,

*"Scrap" is material of sufficient Pu content to make Pu recovery economically attractive to the processor.

which reduces the magnitude of temperature fluctuations and particulate emission rates below those associated with "free-air" incineration. Unburned volatile components, as well as small amounts of entrained particulates, exit the first chamber via an interconnecting port between the two chambers. Excess air and supplemental heat are supplied in this turbulent region to promote combustion. The second chamber provides additional residence time for complete combustion. TRU-bearing solvents are also incinerated in the second chamber. Nominal operating temperatures of 650°C to 870°C in the first chamber and 870°C to 1100°C in the second are maintained by natural-gas-fired burners.

The presence of TRU-contaminated particulates, fission products, and inorganic acids, primarily HCl, in the incinerator off-gas requires a high-efficiency cleanup system before release through the plant stack. The system used (Fig. 4.7) is based on industrial technology and includes the following components:[3]

- A quench column to cool combustion gases
- A venturi scrubber to remove particulates
- A packed-column scrubber to remove acid gases
- A condenser to remove water vapor and lower off-gas temperature
- A mist eliminator
- A reheater to raise the temperature approximately 12°C above the dew point to prevent condensation on downstream components
- A prefilter and two stages of HEPA filters for final particulate removal
- An off-gas blower to maintain flow and keep the process at a pressure below ambient for contamination control
- A plant stack for release of clean off-gas.

The aqueous scrubbing solution is recycled to minimize effluents. Condensate and fresh water makeup are pumped to the top of the packed column. Scrubbing solution from the packed column and quench column is filtered by cartridge-type filters, cooled in a heat exchanger, neutralized, and recycled to the quench column and venturi scrubber. Spent cartridge filters are recycled to the incinerator. Scrubbing solution blowdown is evaporated to dryness. Evaporator overheads are cleaned by the incinerator off-gas system. Incinerator ash is removed by a vacuum system, assayed, mixed with cement, and packaged in DOT 17C drums.

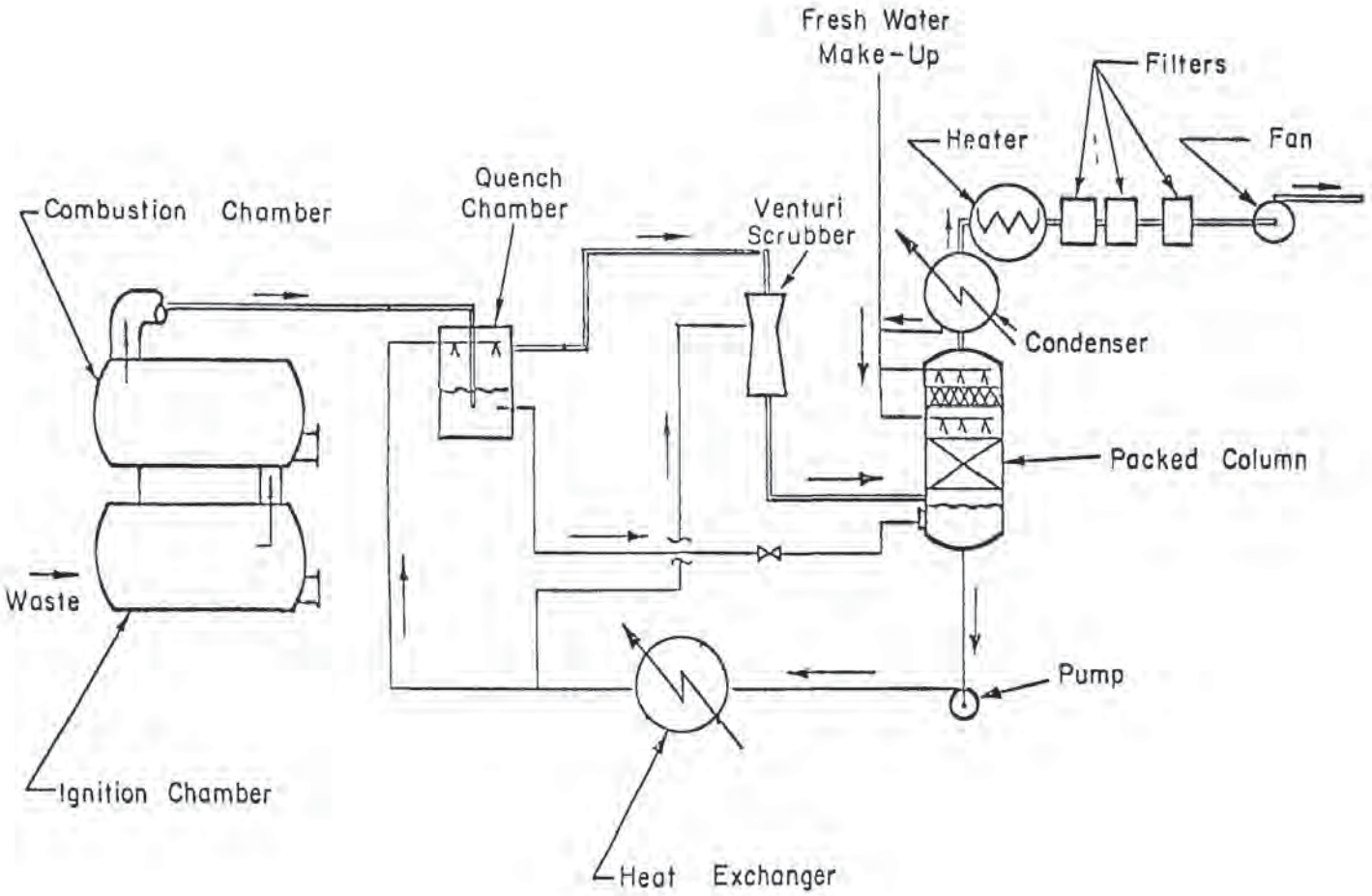


FIGURE 4.7 - CONTROLLED-AIR INCINERATION PROCESS

For many years it has been common practice to use a slurry mixer to incorporate radioactive waste in concrete[7]. The procedure reduces the possibility of radioactive releases during transportation and storage. Ash is mixed with cement at a ratio of 75 wt% ash and 25 wt% cement.

Plutonium content in combustible TRU wastes is low; however, assay of the ash ensures that concentrations will be well within criticality limits. Incineration of combustible wastes from an MOX-FFP[16] produces ash with a concentration of about 1.6 g Pu/kg (assuming about 10:1 weight reduction). Activity concentration in FRP waste is even lower, and the ash from either source can be handled safely in the fixation process.

A suitable container for shipping and retrievable storage is the 55-gallon DOT 17C galvanized steel drum.[17] Corrosion is not a problem with galvanized metal and the 17C drums meet all other criteria specified in Ref. 1, Vol. 3. However, other sizes and materials could be used, depending on the package reliability required.

Evaporation of incinerator scrub blowdown solution will leave a residue high in sodium chloride but low in activity since most of the activity remains with the ash. A suitable shipping and retrievable storage container for the salt residue is a weatherproofed fiber drum inside a DOT 17C container.[18]

4.3.1.3 Bases for Impacts

Since essentially all of the TRU combustible waste will be generated at FRPs, it is assumed that a controlled-air incineration system will be located at these facilities. Releases from the incineration process are compared to site releases.

Release calculations are based on the feed composition shown in Table 4.6, which is within the range projected for TRU combustibles from reprocessing plants.[1]

Table 4.6

NOMINAL FEED COMPOSITION

Component	Weight %
Paper & rags	35
Plastics	
Polyethylene	23
PVC	12
Rubber	30
	100

Performance calculations for particulates are based on a total system decontamination factor (DF) of 1.1×10^{11}. Decontamination factor is system or component input/output. This value is derived from:

- An incinerator DF of 250
- A venturi scrubber DF of 50
- A DF of 9×10^{6} for two stages of HEPA filters[1]

The total DF is conservative for the following reasons:

- The incinerator DF is based on an off-gas particulate loading of 2.2 g/std m^3 with a feed rate of 45 kg/hr. Particulate loadings of 1.25 and 1.46 g/std m^3 at 12% CO_2 were demonstrated in EPA certification tests in a similar unit burning 225 kg/hr of Public Health Service standard waste.[3]
- The venturi DF, under normal operating conditions, will be close to 100.[3]

No credit is taken for particulate removal by the quench column, the packed column scrubber, the condenser, or the mist eliminator.

4.3.1.4 Environmental Impacts

MOX-FFP (Included for Completeness Only)

Using the waste volumes and contamination levels for MOX plants shown in Table 4.5, a specific activity of 0.5 α Ci of Pu, and a total DF of 1.1×10^{11}, the normalized release of TRU materials resulting from incineration is 7.5×10^{-10} Ciα/RRY. This release is approximately 10^4 less than airborne release of radioactive materials from the plant.[16]

Nonradiological airborne effluents resulting from the incinerator operation are shown in Table 4.7.

Table 4.7
NONRADIOLOGICAL EFFLUENTS/RRY
MOX-FFP Waste Incinerator

HCl	0.13 kg
SO_x	0.23 kg
NO_x	0.52 kg
CO	0.25 kg
Particulate	3.6×10^{-9} kg

The sum of these nonradiological effluents is less than 2% of the plant effluents.[16]

Other effluents associated with incinerator operations at the MOX-FFP, normalized to a RRY, are:

- Water discharged to air, 8.8×10^2 gal
- Water discharged to water bodies, 3.3×10^2 gal
- Thermal discharge, 4.3×10^6 Btu

Energy requirements for the process and facility normalized to a RRY are:

- Electrical energy, 3.5×10^{-3} GWe-hr
- Equivalent coal, 1.3 MT
- Natural gas, 4.4×10^{-4} mscf

FRP

Using the waste volumes and contamination levels shown in Table 4.5, a specific activity of 0.5 α Ci/gm Pu, and a system DF of 1.1×10^{11}, the normalized release of TRU materials from an incinerator at an FRP is 1.1×10^{-9} Ciα/RRY, which is 10^5 below the airborne TRU release for the entire plant.[16]

FRP TRU wastes will also be contaminated with fission products. Most of these will form particulates in the incinerator; therefore, the system DF of 1.1×10^{11} applies to most of the fission products. The two exceptions are ruthenium and iodine.

Iodine represents only 1.3×10^{-7}% of the activity in an HLW stream resulting from the dissolution of one-year-old fuel elements (assuming 90% of the iodine volatizes in the dissolver).[1] Since the combustible waste contains only 0.04% of the fission products processed, the total iodine processed by the incinerator represents 5×10^{-11}% of the plant's total activity. If all the iodine in the combustible waste is volatilized, the normalized release would be 1.8×10^{-5} Ci/RRY, which is less than 2×10^{-3}% of the total iodine activity released at the FRP.[16] Furthermore, the incinerator off-gas cleaning system should reduce the iodine release by more than 95%.

Ruthenium constitutes 13% of the total activity in an HLW stream resulting from the dissolution of one-year-old fuel elements.[1] Assuming that the fission-product content of the combustible waste also contains the same fraction, ruthenium could pose a problem during incineration because of its tendency to volatilize under some conditions. However, because the oxygen supply and temperature are limited in the lower incinerator chamber, it is estimated that 0.1% of the ruthenium will volatilize. Furthermore, the off-gas cleaning system should provide a DF of 10^3 for the volatile species.[19] Therefore, the normalized ruthenium release rate due to volatilization would be 1.8×10^{-3} Ci/RRY. If experiments or experience indicate that either the volatilization of ruthenium in the incinerator exceeds 0.1% or the off-gas DF is less than 10^3, the DF can be increased by 10^3 by including ruthenium sorbers in the off-gas cleaning system.[19]

The total normalized fission-product release resulting from the incineration of TRU-contaminated wastes at the FRP is essentially the volatile ruthenium release, or 1.8×10^{-3} Ci/RRY. This value is only 1.2% of the normalized ruthenium release from the other reprocessing activities.[16]

Nonradiological airborne effluents resulting from incinerator operation are shown in Table 4.8.

Table 4.8
NONRADIOLOGICAL EFFLUENTS/RRY
FRP Waste Incinerator

HCl	0.6 kg
SO_x	1.1 kg
NO_x	2.4 kg
CO	1.1 kg
Particulate	1.6×10^{-8} kg

The sum of these nonradiological effluents is approximately 0.02% of the plant effluents.[16]

Other effluents associated with incinerator operations at the FRP, normalized to a RRY, are:

- Water discharged to air, 4.0×10^{3} gal
- Water discharged to water bodies, 1.5×10^{4} gal
- Thermal discharge, 1.9×10^{7} Btu

Energy requirements for the process and facility normalized to a RRY are:

- Electrical energy, 1.6×10^{-2} GWe-hr
- Equivalent coal, 5.8 MT
- Natural gas, 2×10^{-3} mscf

Effluents During Abnormal Occurrences

The design criteria for portions of plutonium facilities[20] require that they be constructed in a manner that will protect the public and operating personnel from hazards associated with normal operations and design-basis accidents including those that might result from catastrophic natural phenomena peculiar to the site. Analyses of consequences of failure, quality assurance reviews during design, and quality control during construction[21] will assure that equipment and structures comply with the applicable safety requirements of Ref. 20. In the Safety Analysis Report for a facility,[22] four areas of review in case of a serious accident or natural catastrophic event are structures, utilities, processes, and process ventilation. The most serious accidents relative to incineration of combustible wastes would be fires and/or explosions.[23]

Structural design features such as massive monolithic reinforced concrete can protect the structure against natural phenomena including earthquakes and tornadoes, and preclude penetration by a severe fire or explosion. Heavily reinforced concrete

construction will also confine a fire, explosion, or nuclear criticality to the room of origin. Utilities and fire-suppression systems using redundancy, self-contained power sources, and emergency reservoirs will be designed to insure continued operation through natural catastrophes. With proper design they can also suppress and confine fire or undesirable releases during emergency shutdown of process systems.

Process safety, including waste-feed characterization, preparation, incineration, and off-gas cleanup, has been intensively addressed in Ref. 23. Explosions in the process line might breach the glovebox and prefilters at the box. However, the volume of the room and ventilation areas will rapidly attenuate the pressure.

Breaching of the ventilation system is thus the occurrence most likely to result in release of radioactive material to the environment. Cooled and cleaned gases from the off-gas treatment system will pass through HEPA filters before release to the atmosphere. Similarly, all air exhausted from the process and support areas will be doubly HEPA-filtered before release. Heat detectors, cool-down sprays, and fire screens in the ventilation ducts will protect the roughing filters and final series of HEPA filters of each system from fire. Alarms, automatic shutdown systems, and isolation dampers will enhance the safety of the ventilation system. The HEPA filters, which are fire resistant, are designed to operate at temperatures up to 300°F and withstand a pressure loading of 10 inches of water gauge across the entire face for short periods without destroying filter integrity.[24] They can survive pulses of 50 milliseconds up to 3 psig.[25,26]

If an explosion breached the entire series of filters, despite pressure attenuation provided by room and ventilation duct and plenum volumes, fire screens, roughing filters, and first-stage HEPA filters, a release to the atmosphere would occur. Assuming failure of automatic transfer to the redundant system, the most severe release would occur during manual transfer to a redundant system, or during manual shutdown of the ventilation blowers. Assuming a manual transfer or shutdown time of one minute, the accidental release would be:

4.5×10^{-4} Ci, Fission Products
1.4×10^{-4} Ciα, TRU

Environmental impacts for the treatment of TRU-contaminated combustible wastes are summarized in Table 4.9.

Table 4.9

SUMMARY OF ENVIRONMENTAL CONSIDERATIONS FOR NORMAL OPERATION OF A NUCLEAR FUEL CYCLE, NORMALIZED TO MODEL LWR ANNUAL FUEL REQUIREMENT, TRU COMBUSTIBLE WASTE

Natural Resource Use	FRP
Land (Acres)	
Temporarily Committed	
Undisturbed area	0
Disturbed area	0.02
Water (millions of gal.)	
Discharged to air	4.0×10^{-3}
Discharged to water bodies	1.5×10^{-2}
Discharged to ground	0
Total Water	1.9×10^{-2}
Fossil Fuel	
Electrical energy (thousand MW-hr.)	1.6×10^{-2}
Equivalent Coal (thousand MT)	5.8×10^{-3}
Natural Gas (million scf)	2.0×10^{-3}
Effluents	
Chemical (MT)	
Gases (MT)	
SO_x	1.1×10^{-3}
NO_x	2.4×10^{-3}
HCl	6.0×10^{-4}
CO	1.1×10^{-3}
Particulates	1.6×10^{-8}
Radiological (curies)	
Fission Products	1.8×10^{-3}
Transuranics, α	1.1×10^{-9}
Thermal (billions of Btu)	1.9×10^{-3}

4.3.2 Treatment of Noncombustible Transuranic Waste

4.3.2.1 Technology Selection

Noncombustible TRU solid wastes comprise fuel-element hulls and hardware, and miscellaneous plutonium-bearing wastes from reprocessing and fuel-fabrication operations.

Fuel-Element Hulls and Hardware

This waste stream is composed of all the undissolved fuel-bundle residues produced in fuel reprocessing. It consists mostly of zircaloy cladding hulls, with lesser amounts of stainless steel, Inconel, and other metals used in end fittings, spacers, etc. The end fittings can be sawed off and stored separately, but for the reference FRP, they are assumed to be combined with sheared fuel-hull segments for joint treatment and disposal.* After acid leaching and rinsing, the spent fuel-bundle residues contain $\leq$ 0.1% undissolved fuel material plus metal activation products formed during irradiation. Fuel-bundle residues generate heat at rates up to 30 watts/ft^3 (1100 W/m^3), depending on fuel burnup and cooling time since discharge. The volume of the combined fuel-bundle residues is about 0.4 m^3/MTHM.[1]

In the reference treatment process, the cladding hulls are immersed in a matrix material to minimize the potential fire hazard from pyrophoric zirconium fines produced during handling operations. Technologies using cement grouts, bitumen, and sand as matrix materials are available.[1] Cement and bitumen have the advantage of almost totally excluding reactive oxygen, as well as immobilizing the waste against formation of fines, and providing increased leach resistance. However, for this study, a sand matrix was chosen because of its fluidity and the assurance that it would fill void spaces.

Other alternatives that have been considered are packaging without treatment, mechanical compaction or densification by melting before packaging, and chemical conversion to more inert waste forms (oxide, sulfates, or fluorides).[2] Of these, only packaging without treatment and dissolution of the zircaloy fraction in aqueous fluoride solutions are considered available technologies.[2] The first has been used without incident at the Nuclear Fuel Services reprocessing plant, but does not provide protection against pyrophoricity of zircaloy fines; the latter process requires corrosive chemicals and produces an aqueous slurry that must be treated further for immobilization.

Failed and Decommissioned Equipment

Failed and decommissioned equipment consists of mostly metallic wastes including process vessels, gloveboxes, piping, pumps, agitators, manipulators, and similar items. Some adsorbent beds and fiberglass filters may be sealed in their process containment vessels and discarded as failed equipment. The quantities of such

*Barnwell Nuclear Fuel Plant Separation Facility, Final Safety Analysis Report, Volume II, Docket 50-332, October 10, 1973, pp. 4-24.

equipment were not specifically identified for the GESMO representative FRP[3], but other published information[4,5,6,10] indicates that anticipated volumes will range from 0.02 to 0.4 m^3/MTHM with a mean of about 0.1 m^3/MTHM.[7] Radioactivities also cover a wide range and are difficult to quantify.

The available technology for the treatment of failed equipment has been developed primarily as a byproduct of maintenance activities in support of radioactive production facilities.[8] The maintenance activities have, in turn, been supported by the development of decontamination processes and the design of equipment to facilitate maintenance operations. It follows that processes for the treatment and disposal of failed equipment will tend to follow the familiar processes already in use. More exotic treatment systems are in various stages of development but are not generally adopted at the present time.[8]

The major alternatives considered were decontamination followed either by direct packaging in specially designed containers or by size reduction to fit smaller standardized DOT containers. The latter alternative was selected for this study because of the assumed transfer of failed equipment to a federal repository as TRU waste and the assumed desire to minimize storage cost in this repository. Further, chopping up of equipment will have the greatest environmental impact at the point of origin.

Ventilation Filters

This waste fraction is composed of roughing filters and HEPA filters used in process off-gas and ventilation ducts to remove aerosols from gas streams. Roughing filters typically are fiberglass pads similar in appearance and purpose to furnace filters. HEPA filters come in several sizes having fire-resistant metal, wood, or press-board frames. The latter frames can be removed and added to combustible trash for incineration. Filters used in primary containment systems can become highly radioactive from trapped fission products and/or plutonium aerosols; e.g., 18 g PuO_2/ft^3 (630 g PuO_2/m^3), or about 0.07% of the PuO_2 throughput in the plant.[10] Those with high fission-product radiation levels (up to 500 mR/hr) will have little or no significant plutonium contamination.[4]

HEPA filters are commonly disposed of by packaging in either 55 or 80-gallon DOT-approved drums, depending on filter size. Future FRPs propose to partially disassemble and compact the filters,[4,10] and this concept was selected for this study. Compaction provides an approximately fourfold volume reduction[10] and reduces the as-generated volume from about 0.06 m^3/MTHM to approximately 0.015 m^3/MTHM for the FRP.

The only other available technology for HEPA filters is packaging without treatment. This can result in gross packaging inefficiency since the size and rectilinear shape of the filters allows only about a 35% packaging efficiency in cylindrical disposal containers.

General Trash

General trash consists of a variety of materials discarded during processing, maintenance operations, and decommissioning. Up to 80 or 90% of this material is combustible, consisting of protective clothing, plastic bags, glovebox gloves, and the like. The noncombustible fraction consists of tools, small failed equipment, ductwork, laboratory glassware, etc. Published estimates of the total quantities involved vary significantly and range from about 0.2 to 1.9 m^3/MTHM for the FRP[11,12] (and from 0.4 to 4.9 m^3/MTHM in the MOX-FFP[13,14]). Most of this waste has relatively low contamination levels, although that generated in plutonium processing areas may contain about 0.04% of the PuO_2 throughput.

The conceptual treatment process chosen for this study includes a sorting step to separate combustible and noncombustible fractions. The former will be burned in a controlled-air incinerator and the ashes immobilized in cement as discussed in Section 4.3.1; the noncombustible fraction will be routed either to the HEPA filter compaction station or the failed-equipment treatment section for disposal. No treatment other than that provided in these previously described treatment facilities is provided for noncombustible trash.

Liquids and Dispersible Solids

This category consists of dispersible wastes in both liquid and solid form. Typical wastes include off-gas scrubber solutions, aqueous raffinates from scrap recovery operations, ion-exchange resins, degraded solvents, silver zeolite beds, UF_6 fluorinator ashes, and adsorbent beds used to remove fluorine and volatile fluorides from UF_6 process off-gas.

The waste treatment process chosen for these wastes is incorporation in cement, which results in a net volume increase of 25 to 100%, depending on the liquid content.[17] However, the volumes of solidified waste projected for the reference facilities are still relatively small, amounting to about 0.15 to 0.2 m^3/MTHM for the FRP.[3,9] Activity levels for most of these wastes are expected to be quite low.

Although degraded solvents and other waste organic liquids are included in this category, only a very small amount of such waste will be solidified with cement (by adding an emulsifying agent, if necessary [15]); the remainder will be incinerated with other combustible wastes, as discussed in Section 4.3.1. The amount of waste organic liquids to be treated is not expected to exceed about 0.01 m^3/MTHM.[16]

There are currently no regulations requiring that dispersible wastes, other than liquids, be solidified before disposal. However, the added safety in handling and transportation afforded by converting these wastes to monolithic solids is believed to warrant the added expense. There are many years of experience in the use of Portland cement to immobilize this category of waste.[17]

Alternative treatment methods include evaporation to dryness (for the liquid fraction) followed by packaging of the dry solids without further treatment, and incorporation of the dispersible wastes in other matrix materials such as urea-formaldehyde or bitumen.[18] The use of a solid matrix material provides added protection against dispersion or leaching if the waste container were to be breached. Urea-formaldehyde has been used extensively to solidify reactor wastes, but there is little experience for TRU wastes. Bitumen is used for TRU wastes in foreign facilities, but potential flammability, particularly with nitrate wastes, has discouraged its use in the United States.

4.3.2.2 Description

We assume that most of the TRU waste treatment and disposal operations will be carried out in a single waste-treatment facility within each FRP. This treatment facility will typically provide waste drum storage, nondestructive assay, and drum loading, sealing, decontamination, palletizing, and other specialized capabilities. Shielding, an overhead bridge crane, manipulators, and a control room are required for the FRP. About 5000 ft^2 is required for the general operation, including temporary drum storage, for each plant. Overall, TRU waste-management operations are expected to utilize about 5 to 10% of FRP floor space.[10,30]

Cladding Hulls

Equipment required to treat and package fuel-bundle residues (cladding hulls and related hardware) includes transfer equipment, a system for blending hulls with sand, and a welding station. The hulls must be air-dried and mixed with dry sand to reduce the pyrophoricity of the zircaloy fines[19,20] and to eliminate pressurization of the closed container by water radiolysis. The hull canister most generally intended for use[21] is sufficiently large to contain the hulls produced from processing 4 MTHM.

Failed Equipment

Operations chargeable to the treatment of failed equipment are decontamination and size reduction. These are basically the same as processes used for normal maintenance operations.

For the reprocessing plant, two treatment facilities are assumed to be provided. One handles highly contaminated failed equipment, while the other handles less radioactive materials. Compaction and selective packaging are used to make the most effective use of space. The sealed packages of waste are ready for shipment offsite or interim storage.

Available packages for failed equipment and other TRU waste include steel drums, steel, wood, or plastic boxes, and plastic drums.[22,23,24] These containers must meet transportation requirements specified in Title 10 CFR 71 and Title 49 CFR 173, and the 20-year retrievable storage criteria of ERDAM 0511.

Treatment of failed equipment will not introduce new effluent control programs. During size reduction, small amounts of contamination may be loosened by mechanical disassembly and will be absorbed by particulate filters in the processing plant ventilation exhaust system.

Ventilation Filters

Conventional drum compactors and presses are available for compacting miscellaneous trash, including HEPA filters. Compactors are used most widely in reactor radwaste management but are also being adapted and used for TRU wastes in ERDA facilities.[25,26] Regardless of the design used, a typical compactor station is relatively small, consisting essentially of a hydraulic system with a ram operating vertically downward, a support plate, frame, and safety enclosure with loading table, vent, and filter system with fan, gauges, and control.[27] A commercial drum compactor has been modified to accept plastic drum liners that may be required for TRU wastes.[25]

Liquids and Dispersible Solids

Several commercially available radwaste cementation systems can be adapted to the solidification of FRP dispersible wastes.[28] The system selected for this study is a drum tumbling system in which a mixing weight and the desired amount of cement are added to the drum in a "cold" area before the drum is transferred to the waste drumming section. The drum is manually or automatically filled with preconditioned waste, usually a slurry, and tumbled to mix the waste with the cement. The overall system requires a control console, a "cold" cement storage and filling station, a waste conditioning and drumming station, and lag storage before shipment or transfer to onsite interim storage. The ash fixation process described in Section 4.3.1 can also be used in a joint facility to immobilize all dispersible wastes.

4.3.2.3 Bases for Impacts

Environmental impacts relative to the reference FRP are discussed in Section 4.1. Plant capacity is assumed to be 2,000 MTHM per year. Wastes from FRP operations will be processed onsite in the main facility. Estimates of quantities of waste requiring treatment are given in Table 4.5. Estimates of airborne releases of radioactivity from waste treatment and packaging operations are given in Table 4.10. The waste treatment facilities have no liquid effluents.[10] The expected radioactivity releases are a very minor fraction (0.001%) of total releases from the main FRP processing operations, as discussed in Section 4.1

The waste treatment facility will require 5 to 10% of the main plant area. Space will be allotted for storage of up to three months' production of waste. The waste treatment facility will be scaled to handle the largest piece of equipment in the plant.

The noncombustible TRU waste treatment facility will be a totally integrated facility yielding waste products in stable form.

Table 4.10

Estimated Airborne Radioactivity Releases

		Annual Release[a]	Release/Accident[b]
FRP	Pu, g	8×10^{-7}	5×10^{-6}
	Pu, αCi	4×10^{-7}	3×10^{-6}
	Fission Products, Ci	8×10^{-5}	---
MOX-FFP[c]	Pu, g	2×10^{-7}	5×10^{-6}
	Pu, αCi	9×10^{-8}	3×10^{-6}

[a]Calculation basis presented in Table 4.11.

[b]Accident frequency not determined but should be less than one per year yielding releases of this magnitude (see Table 4.12 for calculation basis).

[c]Included for completeness.

Reference Environment-Site Characteristics

Less than one acre of land within the site perimeter will be devoted to waste treatment.

Analytical Considerations

As part of the main processing plant, the TRU waste treatment facility will share structural and confinement features. Normal operating conditions should prevail except for accidents.

Normal Operations

Under normal conditions, the impact of this facility upon the environment is expected to be very small. The procedure shown in Table 4.11 was used to estimate the probable release of radioactivity from TRU waste-treatment operations.

Accident Conditions

The main facilities are designed to mitigate risk under severe accident conditions. Confinement provided by the secondary building ventilation system will prevent the release of significant quantities of radioactivity to the environment from such accidents.

The most severe accident analyzed is one in which all plutonium in the waste undergoing treatment is scattered throughout the facility, producing aerosol clouds of plutonium. The plutonium would be assumed to bypass the primary containment system but would largely be contained within the secondary filtration system (10^{-5} fractional release of the aerosol). The calculations in Table 4.12 illustrate the possible quantity of radioactivity released in such an accident. These plutonium concentrations would be reduced rapidly (depending on weather conditions) with distance from point of release, and concentrations near the site boundary would be a quite small fraction of MPC limits.[31]

4.3.2.4 Environmental Impacts

The environmental impact of a treatment facility for noncombustible TRU-contaminated wastes is a small fraction of the total environmental impact of the main process facility.

1. Land Use

There are no land-use effects attributable to the TRU waste-management facility other than those prorated according to the space required in the main plant.

2. Water Use

On a prorated basis, normal operations of the TRU waste-treatment facility will require less than 5% of the water required for the reprocessing plant.

3. Radiological Impact

The radiological impact of the TRU waste-treatment facility is expected to be minimal in comparison to the reprocessing plant. There are no discharges of radioactively contaminated water. Of the volatile FRP isotopes, some tritium is effectively trapped in the cladding hulls, and iodine is solidified in the off-gas treatment wastes. TRU wastes contain little or no krypton. Assuming, on a relative inventory basis, that the fractional release of plutonium and other radioisotopes during waste treatment will be comparable to the fraction released during mainline processing operations, the radiological impact to an individual at the site boundary would not exceed 0.1 to 0.4% of the impact from the main process. For comparison, the dose calculated for a reprocessing plant has been conservatively estimated to be 0.6 millirem annual commitment from transuranics.[29]

Table 4.11

Calculation of Normal Airborne Radioactivity Release

Assumptions:

1. Fractional release of radioactivity as an aerosol entering the ventilation system during treatment = 0.01.

2. Fractional release through the primary ventilation confinement system (assumed to be a roughing filter plus a HEPA filter or equivalent) = 1×10^{-4}.

3. Fractional release through the building secondary confinement system (at least two HEPA filters in series) = 1×10^{-5}.[10]

4. Annual plutonium production for the reference FRP = 26 MT; for the MOX-FFP = 18 MT.

5. Maximum fraction of FRP beta-gamma activity in TRU waste = 0.001; maximum fraction of FRP plutonium activity in TRU waste = 0.003; maximum fraction of MOX-FFP plutonium activity in TRU waste = 0.001.

6. Specific activity of plutonium = 0.5 αCi/g.

7. Total fission-product activity in fuel at time of reprocessing = 4×10^{6} Ci/MTHM.[32]

Calculation:

1. Plutonium release

FRP = $(0.003)\ (26 \times 10^{6}\ g)\ (0.01)\ (10^{-4})\ (10^{-5}) = 7.8 \times 10^{-7}$ g/year
$= 3.9 \times 10^{-7}$ αCi/year

MOX-FFP = $(0.001)\ (18 \times 10^{6})\ (0.01)\ (10^{-4})\ (10^{-5}) = 1.8 \times 10^{-7}$ g/year
$= 9.0 \times 10^{-8}$ αCi/year

2. Fission-product release

FRP = $(0.001)\ (4 \times 10^{6})\ (0.01)\ (10^{-4})\ (10^{-5}) = 4 \times 10^{-8}$ Ci/MTHM
$= 8 \times 10^{-5}$ Ci/year

3. Estimated site boundary concentrations (assumed $\chi/Q = 2.5 \times 10^{-8}$ sec/m^3 for the FRP; 1×10^{-5} sec/m^3 for the MOX-FFP):

FRP α = $\dfrac{(3.9 \times 10^{-7}\ \text{Ci/year})}{(3.15 \times 10^{7}\ \text{sec/year})} \times 2.5 \times 10^{-8}\ \text{sec/m}^3 = 3.1 \times 10^{-22}\ \text{Ci/m}^3$

MOX-FFP α = $\dfrac{(9 \times 10^{-8})\ (1 \times 10^{-5})}{3.15 \times 10^{7}} = 2.9 \times 10^{-20}\ \text{Ci/m}^3$

FRP fission products = $\dfrac{(8 \times 10^{-5})\ (2.5 \times 10^{-8})}{3.15 \times 10^{7}} = 6.3 \times 10^{-20}\ \text{Ci/m}^3$.

For comparison, the MPC_{air} for soluble Pu is 6×10^{-14} Ci/m^3; the MPC_{air} for unidentified radioisotopes is 2×10^{-14} Ci/m^3.[31]

Table 4.12

Calculation of Potential Accidental Airborne Plutonium Release

Assumptions

1. Accident occurs through failure of the primary ventilation system on the HEPA filter compactor while compacting a 2' x 2' x 1' filter containing 15 g PuO_2/ft^3.
2. Fractional PuO_2 release into the secondary confinement area is 0.01.
3. Fractional release through the secondary confinement system is 10^{-5}.
4. The release is exhausted through the secondary confinement system within 20 min.

Calculations

1. Pu released to the atmosphere

   $(15)(4)(0.01)(10^{-5})(\frac{239}{271}) = 5.3 \times 10^{-6}$ grams;

   $(5.3 \times 10^{-6}\ g)(0.5\ \frac{\alpha Ci}{g}) = 2.6 \times 10^{-6}\ \alpha Ci.$

2. Pu release rate $= \frac{2.6 \times 10^{-6}}{(20)(60)} = 2.2 \times 10^{-9}\ \frac{\alpha Ci}{Sec}$

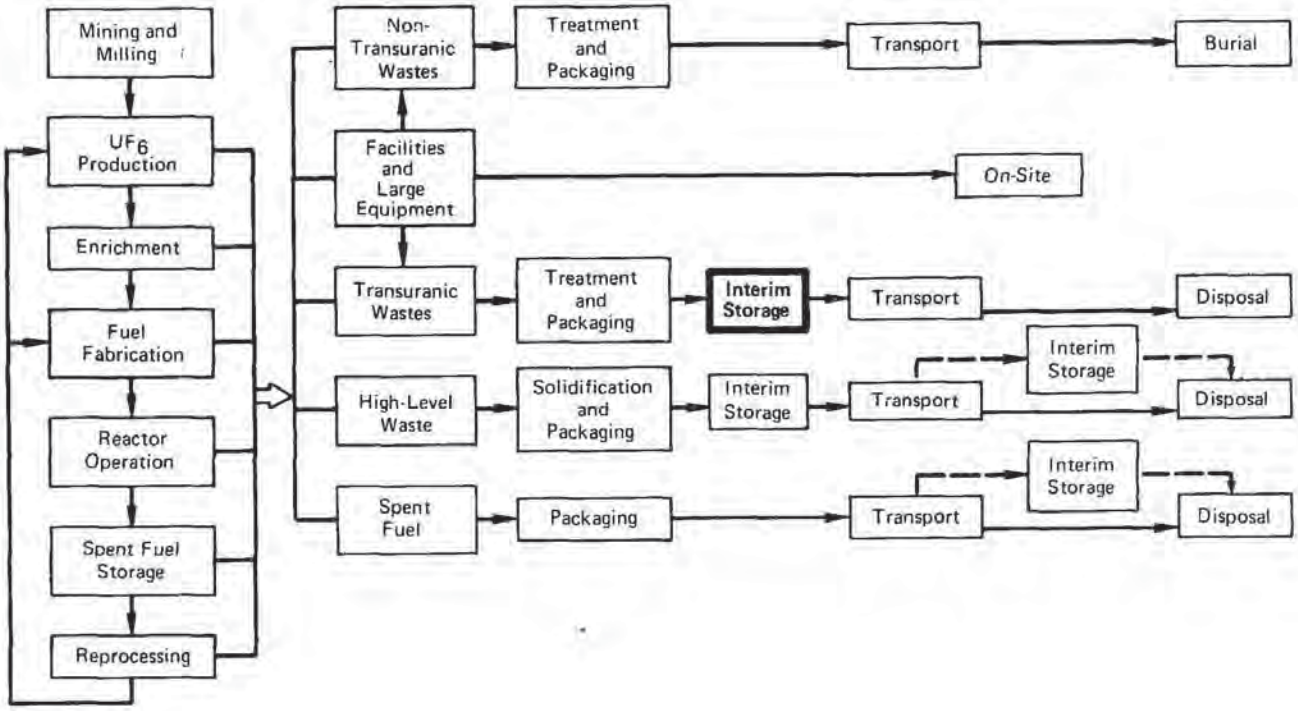
3. Site boundary concentrations:

   For the FRP $\overline{\chi}/Q = 2.5 \times 10^{-8}\ \frac{sec}{m^3}$ (BNFP annual average):[29]
   $(2.5 \times 10^{-8})(2.2 \times 10^{-9}) = 3.5 \times 10^{-17}\ \alpha Ci/m^3.$

   For the MOX-FFP ($\overline{\chi}/Q \simeq 1 \times 10^{-5}\ sec/m^3$, Westinghouse Recycle Fuels Plant),[30] the site boundary concentration would be:

   $(1 \times 10^{-5})\ (2.2 \times 10^{-9}) = 2.2 \times 10^{-14}\ \alpha Ci/m^3.$

   For comparison, the MPC_{air} for insoluble ^{239}Pu is $10^{-12}\ Ci/m^3$.[31]



4.3.3 Interim Storage at the Originating Facility

4.3.3.1 Technology Selection

TRU waste will be shipped to a federal repository for disposal. However, interim storage at the FRP and the MOX-FFP or at a separate designated location will be needed. In this section, it is assumed that interim storage is at the facility that generates the waste. The environmental impacts of waste container transportation are analyzed in Section 4.9 for both onsite and remote interim storage. The duration of interim storage will be less than 20 years; storage packages and conditions, however, will be designed to meet a 20-year retrievability criterion. While a variety of storage container designs might meet this criterion, in this section welded steel containers for cladding hulls and 55-gallon galvanized steel drums are selected as model containers.

Cladding hulls were stored in steel containers in shallow earthen shafts at the Nuclear Fuel Services reprocessing plant before its shutdown.[1] Interim storage systems with improved provision for retrieval of waste containers have been designed.[2,3] In one system, hull containers are stored in the fuel storage basin of the FRP; in the other system, containers are stored in above-surface earthen berms in shafts lined with galvanized steel. The basin storage mode is selected here because of the additional protection it offers against container pressurization that may result in failure from heat generation by radioactive decay and radiolytic hydrogen formation.

The Idaho National Engineering Laboratory (INEL) has had satisfactory experience with earth-covered surface pads for storage of drummed and boxed miscellaneous TRU waste.[4] A conceptual design at one FRP is an adaptation of the INEL mode; steel cargo

containers are used to overpack waste drums for pad storage.[2] A conceptual design at another FRP employs an earth-covered reinforced concrete storage vault equipped with a HEPA-filtered ventilation system.[3] This mode provides considerable extra environmental protection for storage of combustible waste, and is therefore selected for the reference case.

4.3.3.2 Description

Welded stainless-steel containers of cladding hulls and associated hardware are stored in the fuel storage basin of the FRP.[5] Heat from the 3 ft x 8 ft cylinders is removed by circulating pool water. Ventilation air above the basin, and the water in it, are continuously monitored for radioactivity to detect possible leaks. Ten hull containers are generated per RRY. Annual space requirements in storage basins are approximately 250 ft^2, assuming that containers are stored on five-foot centers.

The storage vault for drums of waste is an earth-covered reinforced concrete structure. A structure 200 ft long by 207 ft wide by 21.5 ft high has the capacity for 4600 55-gallon drums. An independent ventilation system draws 14,000 cfm through the facility. Exhaust air is passed through a roughing filter and two HEPA filters before discharge through a 15-foot stack.[6] Annual generation rate (RRY) is 100 drums; about 1000 ft^2 of storage space is required.

4.3.3.3 Bases for Impacts

The high-integrity containers are decontaminated until free of transferable radioactivity before interim storage. Storage locations are monitored for evidence of leaking containers; if such are found, they are repaired or overpacked. Although combustible waste will be incinerated and the ash fixed in concrete (see Sec. 4.3.1) for increased safety during storage and transportation, fire in a waste container is postulated as the worst-case basis for radiological environmental impact. Fire is also postulated as the worst accident for hull containers. Other environmental impacts are relatively small increments of the impacts of the associated FRP.

4.3.3.4 Environmental Impacts

Thermal Effects

The heat generated by a 20-year inventory of cladding hulls in fuel storage basins is less than 10% of the heat generated by fuel in storage awaiting processing. Consequently, the thermal impact of this mode of interim storage is a minor increment of the thermal impact of spent-fuel storage.

Chemical Effluents

Waste water releases from the storage basin attributable to cladding-hull storage are proportional to the respective thermal impacts, and therefore a minor increment of the impact of chemical effluents from spent-fuel storage. Hydrogen generation rate from radiolysis of combustibles or water is less than 800/RRY.

Radiological Impacts

Normal Operations

Handling of these high-integrity containers for interim storage results in a negligible release of radioactivity to the environment; the packages are free of measurable contamination by standard smear techniques, and they have high-integrity seals--welds in hull containers, gasketed lids clamped by a retaining band in drums. Projected release rates for plutonium from storage locations is less than 1 $\mu Ci\alpha$/RRY.[7]

Abnormal Occurrences

Abnormal occurrences or accidents in interim storage operations that offer the greatest potential for offsite consequences are fires in waste containers. Spontaneous fire is a credible accident in the storage of drummed combustible waste, or following breaching and overheating of a welded-hull container.

Fire in the Storage Vault for Drummed Waste[8] -- Spontaneous combustion of waste in a drum is assumed to consume the contents of one drum containing 7 cubic feet of waste over a half-hour period. The burned drum is assumed to have contained the maximum amount of plutonium (200 grams) permitted in a waste package. Being confined by the drum, the fire would not release particulates to the extent ($\sim$5 percent) observed in trash burning. For this analysis, one percent is assumed to be released in a fire which consumed the contents of the drum over a half-hour period.

The filters of the ventilation system (8 sets, each set having one roughing filter followed by two HEPA filters in series) would remain operable during the fire and provide a particulate DF of 10^5. The temperature of the ventilation air would remain well within the structural limits of the HEPA filters. Water sprays, activated by the fire control system, and dilution with 9000 cfm of air would cool and clean combustion gases before their entry into the de-entrainer ahead of the filters.

The estimated quantity of plutonium that would be released from the stack following this accident is 20 μg ($\sim 10^{-4}$ α Ci).[8]

Fire in a Container of Cladding Hulls[9] -- For this analysis it is assumed that the zirconium fire would spread throughout the entire container. Heat from the fire would vaporize some radionuclides in the cladding. Smoke and particulates released during the fire would contain non-volatile radionuclides. The leached cladding is assumed to contain 0.1 percent of the original core material as surface contamination, 100 percent of the tritium as zirconium tritide, and 100 percent of the activation products dispersed in the zirconium metal.

The fire in the leached-cladding container is assumed to burn for approximately eight hours and to consume all of the zirconium. During this time, the quantity of nonvolatile airborne particulates dispersed is estimated at 1 percent of the zirconium inventory.[9] This estimate is based on experiments with plutonium fires.[10] Most of the particles are trapped in the filtration system. If the entire inventory of zirconium

were reduced to particulates and carried from the cell to the ventilation filters, the loading on the deep-bed fiberglass filter would only be approximately 10 percent of its maximum capacity. The filters (deep-bed fiberglass followed by two HEPA filters in series) would remain operable during the fire and provide a particulate DF of 10^5. The estimated quantity of plutonium that would be released from the stack following this accident is 60 μg ($\sim 3 \times 10^{-4}$ α Ci); in addition, an estimated 4×10^3 Ci of tritium and 30 mCi of Co-60 would be released.[9]

Mitigating Techniques -- In both postulated examples of abnormal occurrences, fire in a waste drum or a hull container, the offsite exposure consequences could be mitigated substantially by fixing the hulls in a concrete matrix and by processing the combustibles to ash which is then fixed in a concrete matrix. The fire resistance of this form would decrease by a large factor, perhaps as much as 100, the quantity of transuranics vaporized by fire in these accident situations.

Summary

The environmental impact of the interim storage operation is a relatively minor increment of the impacts attributable to the related major processing facility, whether reprocessing or fuel fabrication. Interim storage is a passive operation in which the principal emphasis is on routine monitoring to verify continuing integrity of the stored waste containers. The active process operations are the straightforward handling operations of actual emplacement and final removal of containers in or from their storage positions, and the continuing supply of ventilation air in the case of drummed waste (~10% of the ventilation air for the FRP) and recirculating cooling water in the case of hull containers (less than 1% of the recirculating cooling water required for stored spent fuel in the FRP storage basin). Environmental impacts are summarized in Table 4.13.

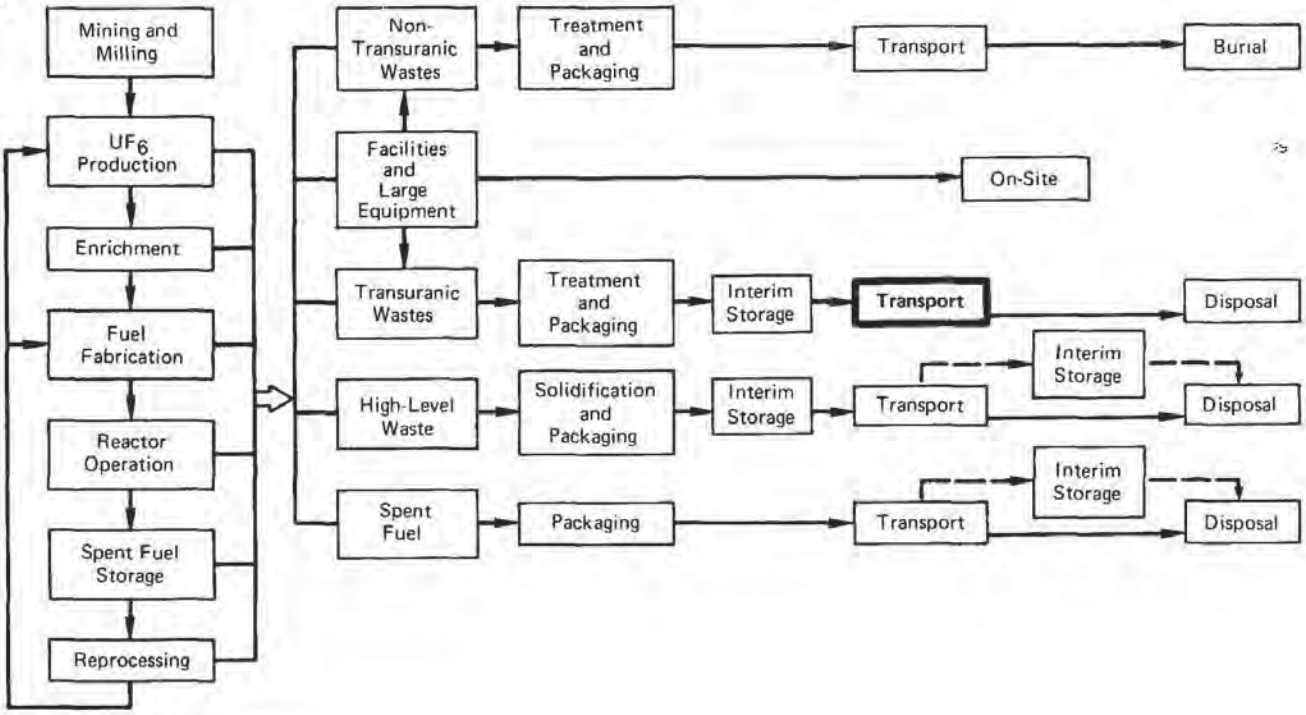
Table 4.13

SUMMARY OF ENVIRONMENTAL IMPACTS; MANAGEMENT OF TRU-CONTAMINATED WASTES (NORMAL OPERATION)
(Normalized to 1 RRY except as noted)

Natural Resource Use	Interim Storage
Land (acres)	
Temporarily committed	
Undisturbed area	-
Disturbed area	0.1
Permanently committed	-
Overburden moved (millions of MT)	0.1
Water (millions of gallons)	
Discharged to air	6×10^{-2}
Discharged to water bodies	-
Discharged to ground	-
Total	6×10^{-2}
Fossil Fuel	
Electrical energy (thousands of MW-hr)	4×10^{-2}
Equivalent coal (thousands of MT)	1.5×10^{-2}
Natural gas (millions of scf)	-
Effluents - Chemical (MT)	
Gases (including entrainment)	
SO_x	-
NO_x	-
Hydrocarbons	-
CO	10^{-4}
Particulates	-
Other Gases	10^{-4} H_2
F^-	-
HCl	-
Liquids	None
Solids	-
Effluents - Radiological (curies)	
Gases (including entrainment)	
Rn-222	-
Ra-226	-
Th-230	-
Uranium	-
C-14	-
Tritium (thousand)	-
Kr-85 (thousands)	-
I-129	-
I-131	-
Fission Products & Transuranics	1×10^{-6} Pu α
Liquids	None

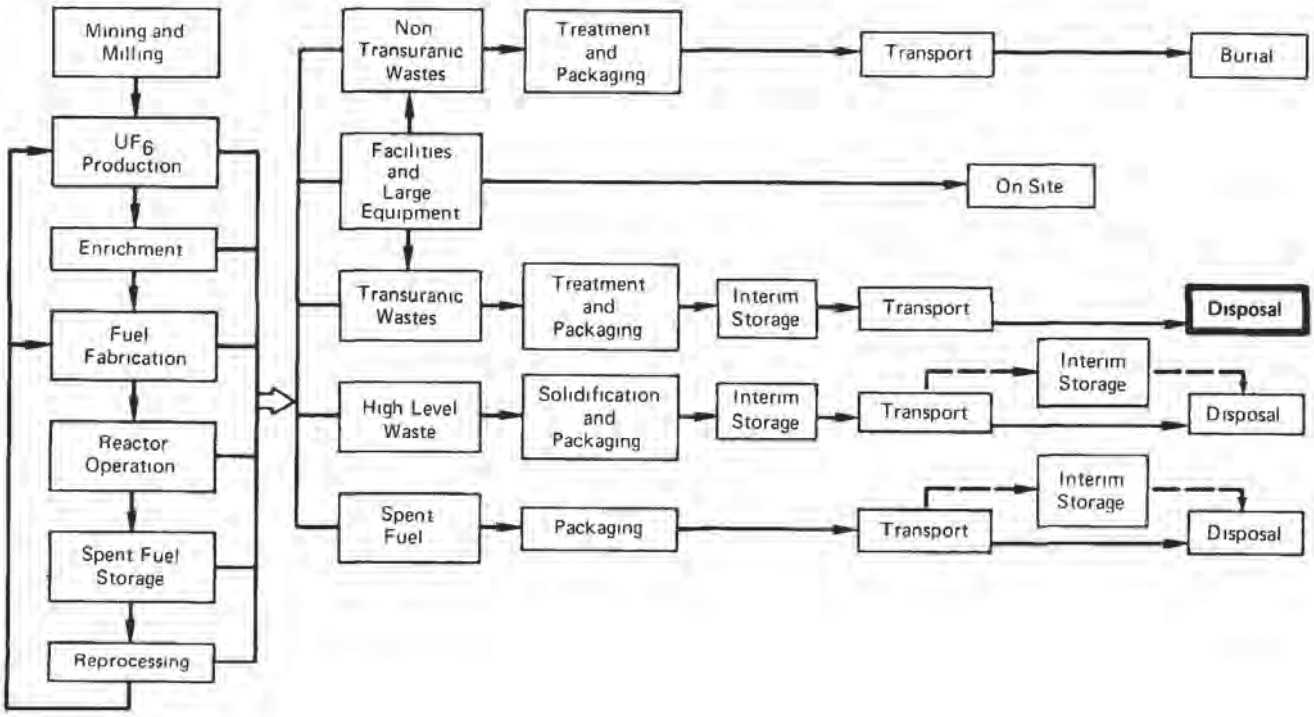
Table 4.13 (cont'd)

Natural Resource Use	Interim Storage
Liquids	
Solids (buried)	
Other than high level	-
TRU	-
High Level Waste	-
Effluents - Thermal (billions of Btu)	0.5



4.3.4 Transportation

Environmental effects resulting from transportation of radioactive wastes to the federal repository are discussed in Section 4.9. Information presented in Section 4.9.4 indicates that the thermal and radiological effects of routine TRU transport are negligible.

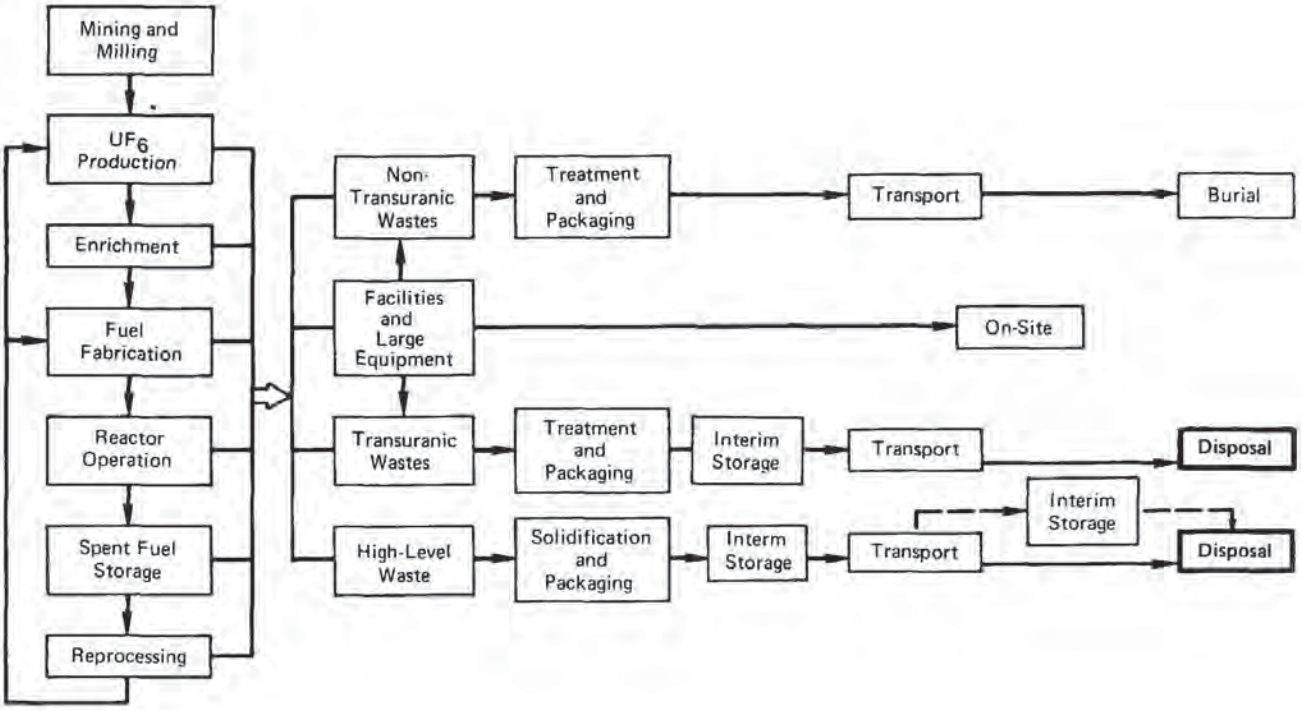


4.3.5 Disposal

Environmental effects resulting from disposal of long-lived radioactive wastes are discussed in Section 4.4.

The geologic isolation system selection for analysis is disposal with conventional mined emplacement in a bedded-salt formation.

Small amounts of radioactive materials are released to the atmosphere during normal operation of the repository. The only source for radioactive release considered significant is the gaseous effluent, which will contain a small number of radioactive particles.



4.4 Disposal of Long-Lived Wastes in a Federal Repository

4.4.1 During Operation

A federal repository would be used to isolate high-level and TRU-contaminated wastes from the biosphere for times related to the biological hazards and the long half-lives of these materials (e.g., the half-life of Pu-239 is ~25,000 years). Management of these wastes before geologic isolation, including interim surface storage, is described in Sections 4.2 and 4.3.

Geologic isolation (the method chosen for discussion here), is the deep emplacement of wastes in natural geologic formations, either for storage or disposal. Storage and disposal are distinguished by the techniques used for waste emplacement and the effect of these techniques on waste retrievability. Geologic disposal would use initial emplacement techniques that are irreversible in an engineering sense and would be selected in the belief that retrieval is not intended.[1] An example is emplacement so that the waste will mix with its surrounding medium to form a relatively homogeneous rock/waste mixture.

In contrast, geologic provisional storage repositories would use emplacement techniques designed to permit engineered waste retrieval.[1] Wastes would be placed in the geologic medium so that their position and configuration would be maintained. Since the repository would be in active use and extensively monitored, the effects of thermal, radiation, and chemical interactions on the stability of the geologic medium could be evaluated, leaving the option open of retrieving the wastes if adverse effects were found. If experiments, tests, and analyses indicated acceptable long-term stability during provisional storage, the repository would be converted to disposal status by backfilling and sealing after the repository was full.

After this conversion, retrievability from a disposal repository is conceivable. Factors influencing retrievability are discussed in Ref. 1.

4.4.1.1 Technology Selection

The geologic isolation system selected for analysis here is disposal with conventional mined emplacement in a bedded-salt formation. Selection of this option is based upon the amount of information available about possible or proposed facilities,[2-5] environmental effects and accident risks during provisional storage,[6,7,8] thermal and radiation effects of HLW on the medium,[9-14] and long-term risks during the disposal period that would result from man-made or natural causes.[15] Most of this information is specific to the salt medium, reflecting the approximately 20 years of analyses and experiments relating to bedded salt as a nuclear waste geologic repository medium.[16] Equivalent information and analyses do not yet exist for other geologic media which could prove equally attractive or even provide advantages over salt for waste disposal.

Other waste isolation options exist. In general, they consist of 1) eliminating portions of the wastes from existence on earth; or 2) storing or disposing of the wastes in other geologic media.

The two alternatives that involve eliminating portions of the wastes from existence on earth are ejection from earth by rocket[17-19] and transmutation.[20-24] The latter involves bombarding the radioactive nuclei with radiation or nuclear particles, thus changing them into other isotopes.

Numerous concepts have been advanced for each alternative. Rocket ejection is made specific mainly by choice of destination, and transmutation concepts are defined by the bombarding particles and their sources. Both alternatives require partitioning[25-28] of the waste.

The three isolation alternatives that involve storage or disposal in geologic media are:

- Ice sheet isolation
- Seabed isolation
- Deep continental geologic isolation

Numerous concepts have also been advanced for each of these alternatives. The concepts are made specific by choice of repository site, waste form, emplacement medium, and emplacement method. The first requires emplacement of solid waste forms on, in, or under glaciers in Antarctica, and is discussed in Refs. 29 and 30. Seabed isolation would involve placement of solid waste forms in sediments or rocks under selected areas of the oceans, as discussed in Refs. 31-34. Deep continental geologic isolation includes a variety of emplacement media and methods and both solid and initially liquid waste forms. Geologic media being considered include bedded salt, dome salt, shales, carbonates, talc, volcanic tuff, metamorphic rocks, and basalt. Geologic descriptions of many of these rock types within the continental U.S. are presented in Ref. 35.

Emplacement methods considered for solid waste include conventional mined cavities,[36,37] solution-mined cavities,[38] a matrix of drilled holes,[39,40] and superdeep holes.[41,42]

Emplacement of liquid wastes by hydraulic fracturing[43-44] and deep-well injection[45] has been considered for selected waste types. Rock-melting concepts with both liquid and solid waste forms are discussed in Refs. 46 and 47. Air or water interim cooling with a solid waste form emplaced in a man-made structure in a geologic formation has been considered.[48] Most of these options are described briefly in Ref. 49, and more extensively in Refs. 50 and 51.

4.4.1.2 Description

The function of a geologic repository is to isolate waste from the biosphere. The method is to emplace the wastes at depth in a tectonically, hydrologically, and mechanically stable medium essentially free of and isolated from mobile groundwater and capable of absorbing radiation and diffusing heat without impairing the integrity of the formation. This stability is the primary element in maintaining waste isolation. In the following sections are discussed important elements of the repository system.

The Site

At present, no repository site has been selected for commercial wastes. ERDA's Office of Waste Isolation is conducting reconnaissance surveys for a bedded-salt repository site for commercial nuclear wastes.[52] The locations of significant rock-salt deposits in the continental United States are shown in Ref. 53. Salt deposits are known to exist in most sections of the country, having been found in 24 of the 50 states.

Site selection criteria are summarized in Refs. 54 and 55 and are more fully developed in Refs. 56 and 57. Topography, tectonics and seismicity, stratigraphy, structure, hydrology, and future geologic events are discussed with reference to such criteria. Site characterization techniques, including mapping, gravity and magnetic measurements, and seismic techniques useful for regional evaluations are given in Ref. 58. This reference then discusses local and site-area evaluation techniques, including the use of exploratory holes, hydrology testing, and the acquisition of rock-mechanics data. The objective of these methods is to locate a stable formation of bedded salt with conditions consistent with the selection criteria referenced above (depth 1000 to 4000 ft; avoidance of areas with high tectonic activity, minimal conflict with valuable natural resources, etc.).

The area of land required for a model bedded-salt repository can best be described with reference to Table 4.14, which conservatively illustrates the type of land-use restrictions that might apply to a bedded-salt repository site.[59,60] More detailed restrictions may be developed on a site-specific basis.

TABLE 4.14

Probable Land-Use Restrictions for a Bedded-Salt Repository Site

Zone #	1	2	3	4
Zone area (acres)	100	1,900	6,000	10,000
Total area of this and all lower numbered zones (acres)	100	2,000	8,000	18,000
Approximate radius (miles)	0.22	1.0	2.0	3.0
Defining use	Surface facilities	Maximum extent of underground repository development	Buffer zone	Buffer zone
Conservative assumptions for activities requiring approval				
Surface	all	all	all	New activities
Underground	all	all	all	all
Probable restriction	For repository use only	No mining or drilling	No mining or deep drilling	Could range from no mining or deep drilling to no solution mining or hydrofracture

The Medium

Bedded-salt deposits are part of evaporite sequences which consist primarily of salt beds interbedded with other sedimentary materials. For example the average lithology from a number of cores totalling 50,811 feet of Paradox Basin strata (in this case salt anticlines in Utah) is given in Ref. 61. The average composition of strata constituting the salt-bearing formation, represented by relative lengths of recovered core, breaks down as follows: salt (72%), shale and siltstone (18%), gypsum and anhydrite (4%), limestone and dolomite (3%), sandstone and conglomerate (2%). Because the bedded salt deposits were formed by evaporation of sea water, they contain not only NaCl, but also several other substances. Ref. 62 gives chemical analyses of 11 rock-salt samples from mines around the world. The NaCl percentage in these "pure" salt beds ranges from 95.8% to 99.1%. Other compounds are $CaSO_4$ (1.6% to trace), $MgCl_2$ (0.4% to 0.02%), and H_2O (1.2% to 0.04%). These chemical and structural impurities vary significantly from location to location. The multicomponent nature of this medium demands site-specific information for detailed discussion. Site selection and evaluation as well as facility design will be based on an understanding of the effects of these inhomogeneities on long-term waste containment.

Plasticity is a more general property of salt. When initially deposited, salt can be as porous and permeable as a coarse sandstone. As the newly formed deposit becomes slightly buried, individual grains yield plastically, deform, and recrystallize. This collapses the pores and squeezes out most of the brine (a small amount is trapped as intracrystalline inclusions, generally about 0.1 mm in diameter), resulting in the formation of a massive, virtually impermeable, polycrystalline rock. Many rock types respond to tectonic stresses and consequent deformation in a brittle manner and develop interconnected fractures which eventually become channels through which groundwater circulates. Salt formations on the other hand, because of their high plasticity, yield and flow while maintaining their massive, impermeable character.

The thermal conductivity of salt is higher than that of many other rocks; this is advantageous in removing decay heat from radioactive waste and in preventing an excessive rise in temperature. Most salt deposits are located in geologically stable regions typified by slow and gradual deformations.

Waste Forms

The HLW form and container chosen for this document are a glass waste form in a metal canister as described in Section 4.2.2. TRU waste forms and packaging are hulls in metal canisters and nonflammable TRU in drums, as described in Sections 4.3.1 and 4.3.2.

Waste and Medium Interactions

Radioactive wastes are sources of heat and radiation and are chemically different from the surrounding geologic media. These effects and combinations of

them may cause changes in the initially stable geologic media which must be considered in repository design and operation. Since these effects cause chemical, thermal, and stress gradients which are potential driving forces for radionuclide migration, they must be understood and characterized before operational safety can be evaluated for either provisional storage or long-term containment.

Radiation doses and predicted and/or observed responses to them are summarized in Ref. 63 and more fully discussed in Refs. 14 and 64. The temperature response of salt to an HLW thermal source, when combined with the thermal stability of the waste, the structural integrity of mined repository areas, heating of the earth's surface, and brine migration effects, limits the emplacement of HLW to about 150 kW/acre.[65,66] The thermal power density associated with cladding-hull wastes is about a factor of 100 lower than with HLW; thermal response of the salt is discussed in Ref. 67. The thermal power density of other TRU waste is about a factor of 10^5 less than HLW, and its thermal effects are discussed in Ref. 68.

Brine-filled cavities, ranging in size down from a few millimeters, occupy about 0.5% of the volume in many bedded-salt deposits. When the cavities are subjected to a temperature gradient, they will move either up or down the gradient because of the different solubilities of salt at different temperatures across the cavity, thus providing a possible means for nuclide transport. A cavity containing only brine will move up the gradient in the direction of the heat source, while a cavity containing some gas can move down the gradient. Gas may already exist in a brine cavity, and additional gas can be generated by radiolytic decomposition of the liquid brine. Brine migration data and analyses are presented in Refs. 69 to 72.

The brine, vapors, and salts around a waste canister will be exposed to high gamma-ray intensities and doses and to high temperatures. Accordingly, they will be subject to changes in composition brought about by radiolysis and thermally induced reactions. Detailed review and analyses[73] of the inflow rates of brine, and on the radiation and thermal chemistries of salt and brines near buried waste, show that the radiolysis products will include, importantly, H_2, O_2, and possibly ClO_3^- and BrO_3^-. Most of the ClO_3^- and BrO_3^- will decompose to halides and O_2 at the high temperatures around a waste can; $Mg(BrO_3)$, if present, may give rise to some Br_2. Nearly all of the H_2, and the accompanying oxidized species, are formed within the migrating brine inclusions by radiations absorbed within the brine and by dissolution of trapped chemically active species from the irradiated solid salt. Some brine is rich in $MgCl_2$ (about 2.3 molar), and hydrolysis of the $MgCl_2$ around a canister will give rise to HCl. Corrosion reactions between the metal canister and water vapor may be sources of H_2.

The possible generation of explosive mixtures of H_2 and O_2 or chlorates or perchlorates, and the potential maximum consequences of their reactions, are discussed in Ref. 74. Firm information on the identities and amounts of radiolytic and thermal reaction products around a waste canister will be furnished by site-specific in-situ experiments to help establish suitable burial and operating procedures.

Metal canisters, packages, and sleeves that may be used in geologic final storage will be subject to oxidation and/or corrosion in amounts dependent upon the metal (or metal alloy) used, the compositions of the gases and/or liquids in contact with the metal, local stresses, and temperature. Inner surfaces of canisters or packages may also be subject to corrosion by the contained wastes and/or by products of decomposing wastes. Experimental work is under way or planned to establish the corrosion characteristics of canister-waste-medium linkages.[75-87]

Nuclide Diffusion

Geologic isolation depends on natural containment by the medium. Hence any possible leakage path must be analyzed. Three radionuclide migration mechanisms other than brine migration (already discussed) are described in the bedded-salt waste isolation literature. They are: (1) isothermal bulk diffusion, (2) isothermal surface diffusion (assumed to represent a worst-case grain-boundary diffusion), and (3) gas transport. Ref. 88 states that "Although no data describing solid-state bulk diffusion appear to have been obtained for systems of direct application to the salt mine, the work which has been done leaves little doubt that this mechanism yields negligible transport, except possibly in certain cases in which radiation may induce space charges within the crystal. Even this effect appears remote when one considers that some of the experimental data were obtained using radiotracer techniques."

Surface diffusion results are summarized in Ref. 89, which states "Until recently, our effort to estimate the significance of surface diffusion (and, by extension, of grain-boundary diffusion) as a mode of isotope migration has been centered on the use of La_2O_3 as an appropriate substitute for plutonium. Our results indicate that... surface diffusion is unlikely to be a source of safety hazards. As a further precaution, however, the experimental work has recently been extended to include an investigation of the behavior of SrO, which will provide a measure of the rate of transport of smaller molecules and, in addition, is of intrinsic interest".

Gas transport through salt has been analyzed.[90] The conclusions are that crushed salt (before reconsolidation) will not significantly retard gas migration, and that site-specific in-situ data are needed for bulk bedded-salt permeability.

Repository Facilities

A probable layout of the facilities of a bedded-salt repository is shown in Figure 4.8. Surface facilities include receiving, inspection, decontamination, repair, and overpack areas for incoming waste shipments, along with administrative and mine support facilities. The mine would consist of conventional room-and-pillar excavation, with specific sections of the mine allocated for use with each waste type. For high-level, cladding, and intermediate TRU wastes, the canisters are inserted in vertical holes bored in the mine floor. A shielding plug is then installed in the hole over the canister to allow subsequent operations and access. For low-level TRU waste that does not require shielding, packages are stored on pallets stacked in the mine rooms. Process equipment is used to decontaminate any radioactive liquid wastes

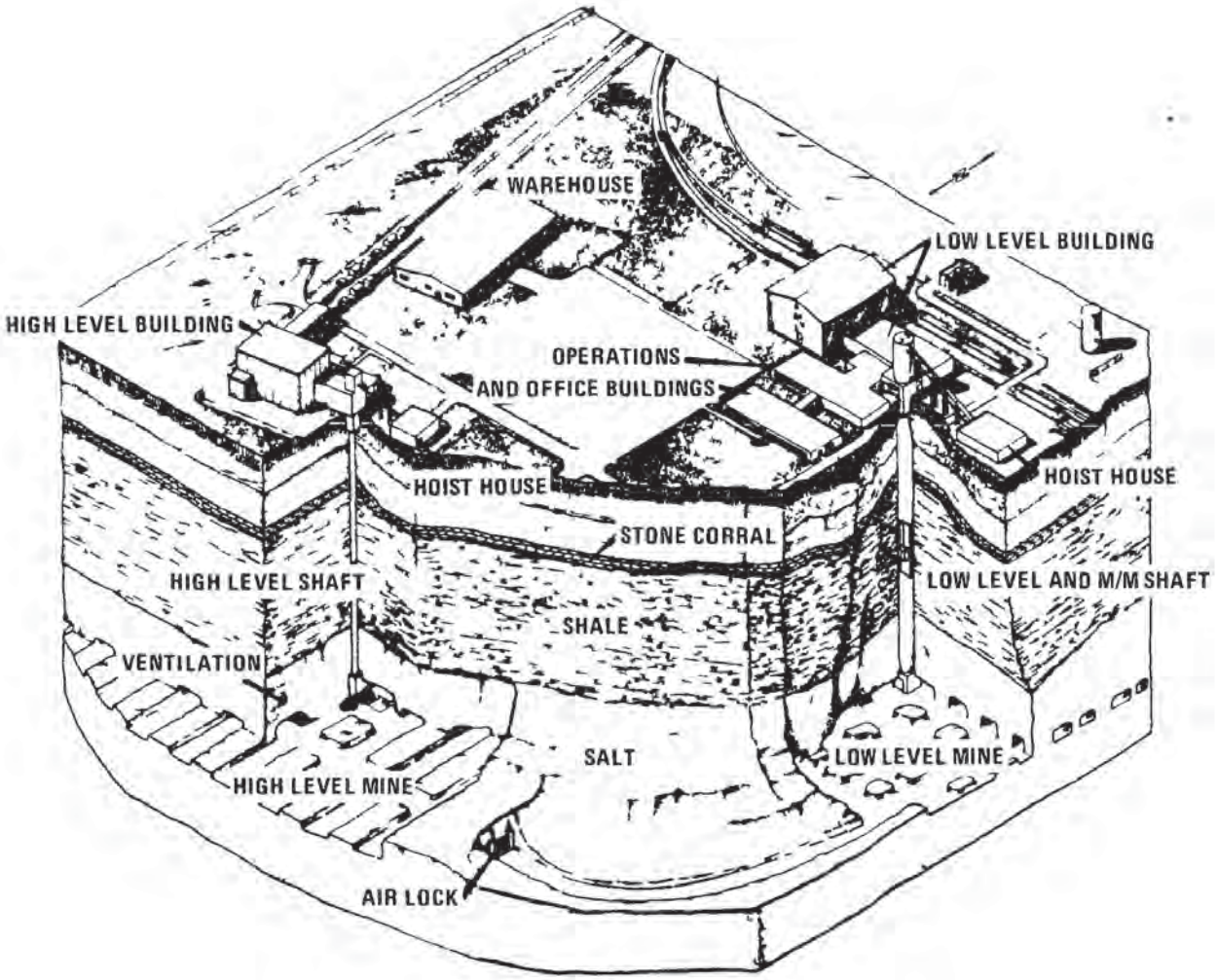


FIGURE 4.8 - PROBABLE LAYOUT OF BEDDED SALT REPOSITORY

generated in repository operation and to reduce the wastes to a solid form suitable for disposal. There are no intentional discharges of liquids containing radioactive materials from the repository. Refs. 91 and 92 provide a summary and a more detailed view of probable repository facilities.

For purposes of normalizing repository impacts to a RRY, the repository is assumed to consist of 1250 acres for HLW (or spent fuel - see Sec. 4.6), 100 acres for cladding hulls, and 200 acres for other TRU wastes. The rest of the 2000 acres of underground development is shafts, corridors, barriers, pillars, etc. This repository will handle the wastes from 4000 RRY's of any of the fuel cycles. The 1250 acres is calculated by using 4000 RRY, 35 MTHM/RRY, 1.31 kW/MTHM (for 10-year-old waste, this is the maximum for the three fuel cycles[94] and 150 kW/acre. The TRU areas are conservatively based on the volumes from Table 4.4(1) and the thermal output of cladding hulls. Provisional storage time of the repository is projected to be 20 to 30 years,[95] and utility requirements are estimated (upper bound) to be:[95]

- Electric power (average full-time usage) - 8 MW
- Water discharged to air - 25 gpm
- Water discharged to water bodies - none
- Water discharged to ground - 200 gpm
- Natural gas* (average full-time usage) - 50,000 ft^3/hr

Conversion to Permanent Disposal

If experiments, tests, and analyses performed during provisional storage indicate acceptable stability for the time and space domains needed, the repository would be converted to disposal status. This conversion would probably involve decontaminating and decommissioning of above-ground facilities, backfilling underground tunnels and rooms, and plugging all boreholes and shafts. After backfilling, the plasticity of the bedded salt would permit a gradual closure of excavated openings, with compaction, reconsolidation, and recrystallization of the salt backfill in the rooms. When this process is completed, the wastes would be contained in a massive and solid salt formation. The current status of borehole plugging studies, including hydrothermal, molten salt, and natural-earth plugs, is discussed in Ref. 96. Rock-mechanics analyses of reconsolidation are discussed in Ref. 97.

4.4.1.3 Operational Impacts

Resource Use

Land - the following commitments are calculated on the assumption of a 4000 RRY repository. Zones 1-3 (Table 4.14) are conservatively assumed to be permanently committed surface areas (2.0 acres per RRY) and the underground resources of zones 1-4 (4.5 acres per RRY) are assumed to be committed. It is possible that after conversion to permanent disposal the surface land-use restrictions of Zones 1-3 could be removed, and some mining or deep drilling might be allowed in Zone 4.

*Other fuels, in amounts required to produce the same heating value, may be used to conform with established policies for energy conservation.

Water - the following estimates are calculated assuming a 30-year provisional storage period and a 4000 RRY repository capacity:

- Water discharged to air = 100,000 gallons per RRY
- Water discharged to water bodies = none
- Water discharged to ground = 800,000 gallons per RRY

Fossil Fuels - Similarly calculated, commitments are:

- Electrical energy = 530 MW-hr per RRY
- Natural gas* = 3.3 millions scf per RRY

Salt - Exclusive of naturally occurring brines such as the seas, salt lakes, and salt springs, rock salt reserves in the United States are astronomical; they have been estimated at 60,000 billion tons. At 100 times the present consumption and at a 50-percent recovery, these reserves are sufficient for 100,000 years.[98] Assuming a Zone 4 area, and a salt thickness of 0.2 km, the salt resource commitment is $\sim 1 \times 10^{7}$ MT/RRY, or $\sim 2 \times 10^{-7}$ of the reserves/RRY. Sites will be selected to eliminate or minimize interference with other resources (e.g., potash, oil and gas, other minerals); therefore we expect no other resource commitments.

Effluents

Small quantities of radioactive and other materials are released to the atmosphere as a result of normal operations of the provisional repository. The only significant source for radioactive release is the gaseous effluent which will contain small amounts of radioactive particles that have passed through multiple HEPA filters. Solid radioactive wastes generated at the repository are packaged and buried in the repository. No liquid wastes are intentionally released. Aqueous wastes are recycled and any excess water will be evaporated and released to the atmosphere. Ref. 99 discusses the sources of gaseous effluents and their airborne concentrations in various facilities. Ref. 100 shows the average annual discharge rate of radioactive and other materials from a repository which yearly accepts the waste from 208 reference reactors. (Ref. 101 indicates that Table IV H-16 represents the effluents from a waste flow equivalent to 7283 MT/yr. This figure, combined with 35 MT/RRY, is indicated above.) These effluent calculations are based on published estimates from the proposed Lyons repository.

On an average annual basis the repository described in Section 4.4.1.2 accepts the waste from 4000 RRY/30Yr=133 reference reactors. Hence Table IV H-16 represents a 56% overestimate of average annual rates of discharge. We will use the Table IV H-16 values of average annual rate of discharge as presented in Table 4.15.

Table 4.16 normalizes the effluents to a RRY basis, by multiplying Table 4.15 entries by 30 years and dividing by the repository capacity of 4000 RRY.

*Other fuels, in amounts required to produce the same heating value, may be used to conform with established policies for energy conservation.

For purposes of Table 2.10, we must divide entrained solids in gaseous effluent streams into fission products and transuranic components. The curies in Ref. 103 representing TRU elements have been divided by the total curies indicated in the same table for each fuel cycle mode. This fraction of the HLW particles of Table 4.16 of this document has been added to the TRU waste particle effluent from 4.16 and appears in Table 4.17 as TRU effluent for each fuel cycle. The rest of the HLW particle contribution appears as fission-product effluent.

The 50-year individual and population dose commitments resulting from operational releases of 1 RRY of waste are given in Table 4.18, which uses the assumptions described in Ref. 102. RRY normalizing is again done by multiplying by the 30-year operational time and dividing by the 4000 RRY repository capacity.

Disposal of Excess Salt

Possible methods for disposal of mined salt that is not used for backfill[104] include sale for commercial use, backfill of abandoned mines, disposal at sea, and dissolution and injection in a low-brine aquifer.

Operational Accidents

Site-independent operational accidents are similar to possible accidents at facilities that provide interim storage. A HLW canister drop is discussed in Section 4.2.5. Cladding-hull and low-gamma TRU fire accidents are discussed in Section 4.3.3. Site-dependent operational accidents, such as mine flooding, require site-dependent data for analysis, and these are not currently available. Site selection criteria and facility design will minimize both the probability and consequences of such an accident.

Long-Term Thermal Effects

The integrated long-term energy output from HLW has been calculated[105] to be 56,400 watt-years/MTHM between 10 years and 10,000 years out of the reactor core. Assuming 35 MTHM/RRY, the heat transferred to the geologic formation during this time will be 59×10^9 Btu/RRY.

Figure 3.6 of Ref. 106 indicates the long-term (up to 5,000 years) thermal response as a function of depth for a particular mine depth and thermal model. The projected complete inventory of waste for a repository might, over an estimated 800 years, produce a peak heat flux at the ground surface of no more than six or seven times the natural geothermal flux. The associated maximum increase in surface temperature has been estimated to be less than 0.1°F, which would have no perceptible effect on surface climate, meteorology, or ecology. By comparison, the average annual solar flux reaching the ground is about 10,000 times greater than the geothermal flux.[107]

Long-Term Surface Elevation Changes

Preliminary calculations for surface displacement associated with thermal expansion and mine closure are given in Ref. 108.

There would be an extremely slow subsidence due to plastic deformation of the mineral to close the mined cavities, modified at first by thermal expansion and later by thermal contraction. It is estimated that over several thousand years, this process would result in the development of a very broad, uniform, and shallow (about 3 1/2 ft) subsidence over the general area above the storage repository. The rate of this subsidence is estimated to approximate that of the ground above a producing mineral mine of equal size, but its magnitude is projected to be about 1/2 as much as that of a mine because of backfilling of rooms and tunnels at the time of closing and sealing of the repository. As part of site selection procedures, site-specific analyses will be made of the effect of this slow subsidence on the integrity of the overlying rock.

Table 4.15

ESTIMATED RELEASE RATES AND AVERAGE ANNUAL OFFSITE CONCENTRATIONS OF RADIOACTIVE AND OTHER MATERIALS RESULTING FROM NORMAL REPOSITORY OPERATION

Material	Average Annual Rate of Release to Atmosphere	Average Annual Offsite Concentrations Resulting From Repository Effluents*	Percent of Applicable Standards for Exposure of the Public**
High-level waste particles	0.007 Ci/yr	2×10^{-15} Ci/m^3	0.02
TRU waste particles	0.03 μCi/yr	1×10^{-20} Ci/m^3	0.0001
^{85}Kr (spontaneous fission)	0.014 Ci/yr	4×10^{-15} Ci/m^3	0.000004
^{3}H (spontaneous fission)	0.0009 Ci/yr	3×10^{-16} Ci/m^3	0.0000004
^{222}Rn (natural sources)	0.9 Ci/yr	2×10^{-13} Ci/m^3	0.02
^{220}Rn (natural sources)	0.04 Ci/yr	1×10^{-14} Ci/m^3	0.0003
H_2 (corrosion, radiolysis, electrolysis)	4 scfm	0.02 ppm	0.0002
He (alpha decay)	0.001 scfm	0.000004 ppm	4×10^{-8}
HCl (brine decomposition)	0.07 scfm	0.0003 ppm	0.006
CO_2*** (diesel exhaust)	50 scfm	0.2 ppm	0.004
CO (diesel exhaust)	0.05 scfm	0.0002 ppm	0.0004
NO_2 (diesel exhaust)	0.05 scfm	0.0002 ppm	0.004
SO_2 (diesel exhaust)	0.03 scfm	0.0001 ppm	0.002
CH_2O (diesel exhaust)	0.0007 scfm	0.000003 ppm	0.00003
Soot (diesel exhaust)	2 lb/yr	0.1 μg/m^3	0.007
Salt particles	5 lb/yr	0.2 μg/m^3	0.001

*These are the maximum concentrations which result at the fencepost of the site.

**Based on one-third of limits in 10 CFR Part 20, Table II, Column 1 for radionuclides and threshold limit values (American Conference of Governmental Industrial Hygienists) for other materials.

***CO_2 is a normal constituent of air and is not considered a pollutant.

Table 4.16

ESTIMATED EFFLUENTS
PER RRY FROM NORMAL REPOSITORY OPERATION

Material	Operational Release Per RRY
High-level waste particles	5.3×10^{-5} Ci
TRU waste particles	2.2×10^{-10} Ci
^{85}Kr (spontaneous fission)	1.1×10^{-4} Ci
^{3}H (spontaneous fission)	6.8×10^{-6} Ci
^{222}Rn (natural sources)	6.8×10^{-3} Ci
^{220}Rn (natural sources)	3.0×10^{-4} Ci
H_2 (corrosion, radiolysis, electrolysis)	4.0×10^{-2} MT
He (alpha decay)	2.0×10^{-5} MT
HCl (brine decomposition)	1.3×10^{-2} MT
CO_2* (diesel exhaust)	$1.1 \times 10^{+1}$ MT
CO (diesel exhaust)	7.0×10^{-3} MT
NO_2 (diesel exhaust)	1.1×10^{-2} MT
SO_2 (diesel exhaust)	9.6×10^{-3} MT
CH_2O (diesel exhaust)	1.1×10^{-4} MT
Soot (diesel exhaust)	7.0×10^{-6} MT
Salt particles	1.7×10^{-5} MT

*CO_2 is a normal constituent of air and is not considered a pollutant.

Table 4.17

HIGH-LEVEL AND TRU WASTE PARTICLES

U-Only Recycle (Normal Operation)

Fission Products	4.2×10^{-5} Ci/RRY
Transuranics	1.1×10^{-5} Ci/RRY

Table 4.18*

SUMMARY OF 50-YEAR DOSE COMMITMENTS PER RRY TO AN INDIVIDUAL** AND A POPULATION*** DUE TO NORMAL OPERATIONAL REPOSITORY

	(U-Recycle Only)	
	Individual (rem/RRY)	Population (person-rem/RRY)
Total Body	3.6×10^{-7}****	9.0×10^{-4}****
G.I. Tract	2.0×10^{-7}	2.0×10^{-4}
Bone	2.5×10^{-6}	4.4×10^{-3}
Liver	4.6×10^{-7}	9.0×10^{-4}
Kidney	1.1×10^{-6}	3.2×10^{-3}
Thyroid	1.3×10^{-7}	1.1×10^{-4}
Lung	3.2×10^{-7}	3.9×10^{-4}
Skin	1.3×10^{-7}	1.1×10^{-4}

* Source: Ref. 102.

** Dose to individual is at site boundary 500 meters from point of release.

*** Dose to U.S. population integrated from 7.5 persons per sq mi in the West to 160 persons per sq mi in the East, over a distance of 2000 miles.

**** The 50-year dose commitment due to natural background to an individual is 5 rem. To the total U.S. population (2×10^8 persons) it is 1×10^9 person-rem.

4.4.2 Long-term Risks from Radioactive Wastes in a Geologic Repository

4.4.2.1 Approach to Assessment of Long-term Risk

Risk is commonly perceived as a range of possible consequences of an action and the likelihood (probability) that each type of consequence might occur. When driving an automobile, the possible adverse consequences range from none to a fatal accident. Experience indicates that the probability of no adverse consequence is high and the probability of a fatal accident is low, largely as a result of regulations and safety features built into cars and highways. The probability of adverse consequences can be greatly increased, however, by abnormal circumstances such as steering system failure.

Similar concepts apply to risks from disposal of radioactive wastes in geologic formations. There will be a range of possible consequences and attendant probabilities, and action will be taken within current regulatory functions and technology (e.g., via licensing criteria and waste forms) to mitigate the probability of adverse events. However, time periods of concern are too long for development of an experience data base sufficient to verify results of analytical estimates; demonstrating the risks associated with this technology is not possible.

Although rigorous verification of long-term risks for waste repositories is not possible, a methodology exists, and has been used, for estimating these risks. Events that could produce or lead to repository failure can be identified, probabilities of the events estimated, and consequences calculated. The procedure requires data and extrapolation of past experience with probabilities of the several phenomena. The degree of confidence in the results depends on the availability and applicability of the data and the validity of the extrapolations.

Possible Events Causing Repository Failure

Geologic waste repositories will be sited, designed, operated, and regulated to minimize possibilities of long-term loss of waste isolation. However, geologic media and natural processes change with time, and emplacement of wastes--which introduces thermal and radiation sources to the medium--may effect the changes. In addition, human activity and natural events independent of the repository may produce changes.

Types of events that can directly or indirectly produce repository failure can be categorized as follows:

Group 1 - Natural rapid events uninfluenced by humans; e.g., meteorite impact that breaches repository containment;

Group 2 - Natural geologic events or processes independent of the existence of the repository; e.g., faulting and erosion;

Group 3 - Geologic events or processes caused by the repository; e.g. events or processes stimulated by the radiation source, thermal source and gradients, etc.; and

Group 4 - Human actions directly or indirectly effecting the repository, e.g., acts of war or sabotage, mineral resource exploration, reservoir construction, etc.

An analysis of a Group 1 event has been performed by Claiborne and Gera.[1] For an assumed waste emplacement depth of 600 meters, they calculated that a meteorite 2 km in diameter would be the minimum size that could breach the overburden. They then analyzed geological evidence of meteorite impacts and concluded that the probability of such a meteor hitting a given square kilometer of the earth's surface is 2×10^{-14} per year (once every 50 trillion years).

An example of a probability estimate for a Group 2 event is the calculation of the probability of faulting through the Delaware Basin in Southeast New Mexico.[1] It was concluded that the probability of a fault intersecting a 8 km^2 repository in that basin is 4×10^{-11} per year (once in 25 billion years). A review of other natural geologic processes relevant to waste disposal is presented in Ref. 3, Sections 5 and 6.

Group 3 events or processes (those stimulated by the presence of wastes) have been discussed in Ref. 4 (Vol. 1, Sec. 3), but probabilities have not been estimated. Most of the work on these mechanisms has been concentrated on potential consequence evaluation. For example, Ref. 5, Section 9.4 concludes that "we are confident that energy storage [due to radiation damage] under the conditions that will prevail in the Repository does not carry any serious implications." Ref. 6, p. viii, in summarizing radiolysis and hydrolysis in brines, concludes that "no possibly-serious problems arising from radiolytic or thermal effects in the repository were recognized which couldn't be counteracted by some modification of the design or operation of the repository. However, the possible effects have not been completely evaluated." Both references indicate that further work is planned. Other studies have been performed.[7-10]

Site selection criteria and site evaluation requirements are designed to minimize the probabilities and consequences of geological events or processes that would occur independent of the repository (Group 2), and waste/media interactions and facility stability studies are designed to furnish data for the understanding and minimization of the probabilities and consequences of geologic events or processes that result from the presence of the repository (Group 3).

Group 4 (human intervention) events such as nuclear war and intrusion by drilling have been investigated. Ref. 1, p. 11, indicates that for an assumed 600-meter repository depth, a surface burst of a 50-megaton nuclear weapon would not breach the geologic containment. Ref. 4, Vol. 1, p. 3.34 ff, discusses man's future activities and probabilities of inadvertent contact with waste in a geologic repository. It calculates the probability of violation of repository integrity as a result of random wildcat drilling to be 4×10^{-4} per year (once in 2500 years).

The latter result, based as it is on past data and an assumption of random action, is a first-order estimate of probabilities of inadvertent intrusion by human activity. Circumstances could increase (nonrandom drilling for resources at the site) or decrease (controlled access) the probability.

In general, a range of probabilities is possible, and the values of the probabilities will depend on the scenarios assumed. Complete analysis will also require some assessment of the probabilities of occurrence of the scenarios themselves and assessment of the probability and consequences of nuclide release for each scenario (consequences were not evaluated in the study cited above). In-depth studies in this sector have not been performed.

4.4.2.2 Consequences of Waste Repository Failure

Assessment of the consequences of a waste release requires description of when and where an individual will come into contact with a particular concentration and type of radioactive material. This contact may be a result of physical mechanism (resuspension and inhalation of small waste particles, transport of waste material in water used for consumption, external radiation from the waste in the immediate surroundings of the individual, etc.) or biological mechanisms (the uptake of waste from the soil by plants in the food chain of humans, concentration of waste constituents in marine life leading to human consumption, etc.).

Assessment of long-term consequences should consider future human habits and demography. Since there is no means of accurately determining distant future societal habits and demographic data, one approach is to assume that societal habits and population distributions will not change much from those of today. Alternatively, concentrations of waste materials in the lithosphere, hydrosphere, and atmosphere can be projected for some future time, and arbitrary future societal scenarios can be superimposed on these distributions. As a reference case, and for perspective, the present society, with its habits and population distributions, can be and has been used.

To date there have been two general approaches to the assessment of consequences of a waste release from a repository. One approach is through the use of a "hazard index" and the other is to model as realistically as possible the movement of waste material to and through the human environment. These approaches are discussed below.

4.4.2.2.1 Hazard Indices as Risk Indicators

The hazards to human health implicit in specific concentrations of radionuclides are often quantified by an arbitrary index called the radiotoxic hazard index (RHI). RHI's may be defined for the various pathways by which radionuclides contact man; a typical RHI is defined for the ingestion of drinking water, and will serve as an example. For a radionuclide of type i,

$$(RHI)i \equiv \frac{Qi}{(MPCw)i}$$

where Qi is the total, time-dependent radioactivity in curies of the ith nuclide in the radioactive material to be considered and (MPCw)i is the maximum permissible

concentration of nuclide i in drinking water, measured in curies/m^3. A fair working definition of (MPCw)i is that it is the permissible concentration of the radionuclide in drinking water that may be consumed at normal rates for 50 years. RHI is, therefore, a measure of the amount of water required to dilute Qi to permissible concentrations. The total RHI of a specific quantity and mixture of radionuclides is simply taken as the sum of (RHI)i's. Examples of such hazard indices for the waste from one ton of spent LWR fuel are given in Figures 4.9 to 4.11,[11] which also illustrate the effect of radioactive decay on hazard magnitudes.

The RHI is only a measure of potential hazard due to existence of the radioactivity; i.e., it does not take into account dynamic phenomena such as different relative rates of migration of various nuclides through soils and different biosphere pathways that can significantly affect actual risks. The RHI applies to situations such as a perfectly mixed body of water or a solid material not subject to dynamic effects. The latter corresponds to an undisturbed repository, and it is to this situation that the RHI has been applied.[7-10]

The RHI for uranium ore has been calculated and compared to that of HLW and other radioactive materials (Figs. 4.9 to 4.11).[11] The amount of uranium ore used for the comparison is equivalent in mass to that of HLW from one metric ton of fuel plus associated geologic medium. The comparison given in Figure 4.11[11] indicates that after 10^3 years the hazard potential of wastes in a repository is less than that of an equivalent amount of uranium ore.

Assessment of Long-Term Consequences by Use of Transport Models

Some studies of potential long-term consequences have taken the approach of assuming the waste repository fails at a particular time and in a particular way, and then estimating consequences by using models that predict transport of radioactivity through appropriate geosphere and biosphere pathways. The approach contains no information on probability of repository failure. Conditions of this assumption (time and mode of failure) are highly important to the results obtained because they determine the quantities and types of radioactivity that can have consequences.

Past modeling assessments of the consequences of releases from a waste repository[1,2,4,15-17] exhibit considerable variation in the level of sophistication. Those considered most realistic conservatively project radiation doses to man in the range of normal background radiation; potential adverse consequences are mitigated by dilution of waste material within the geologic medium and holdup in surrounding media.

One assessment assumed that the contents* of the repository are released at a specified rate and a specific time after emplacement to an underground water stream which flows at a constant velocity directly through the soil column to a surface water body.[2] All radionuclides were assumed to be soluble, to remain in the same chemical

*The repository is assumed to contain all the high-level wastes generated by the commercial U.S. nuclear power industry through the year 2000.

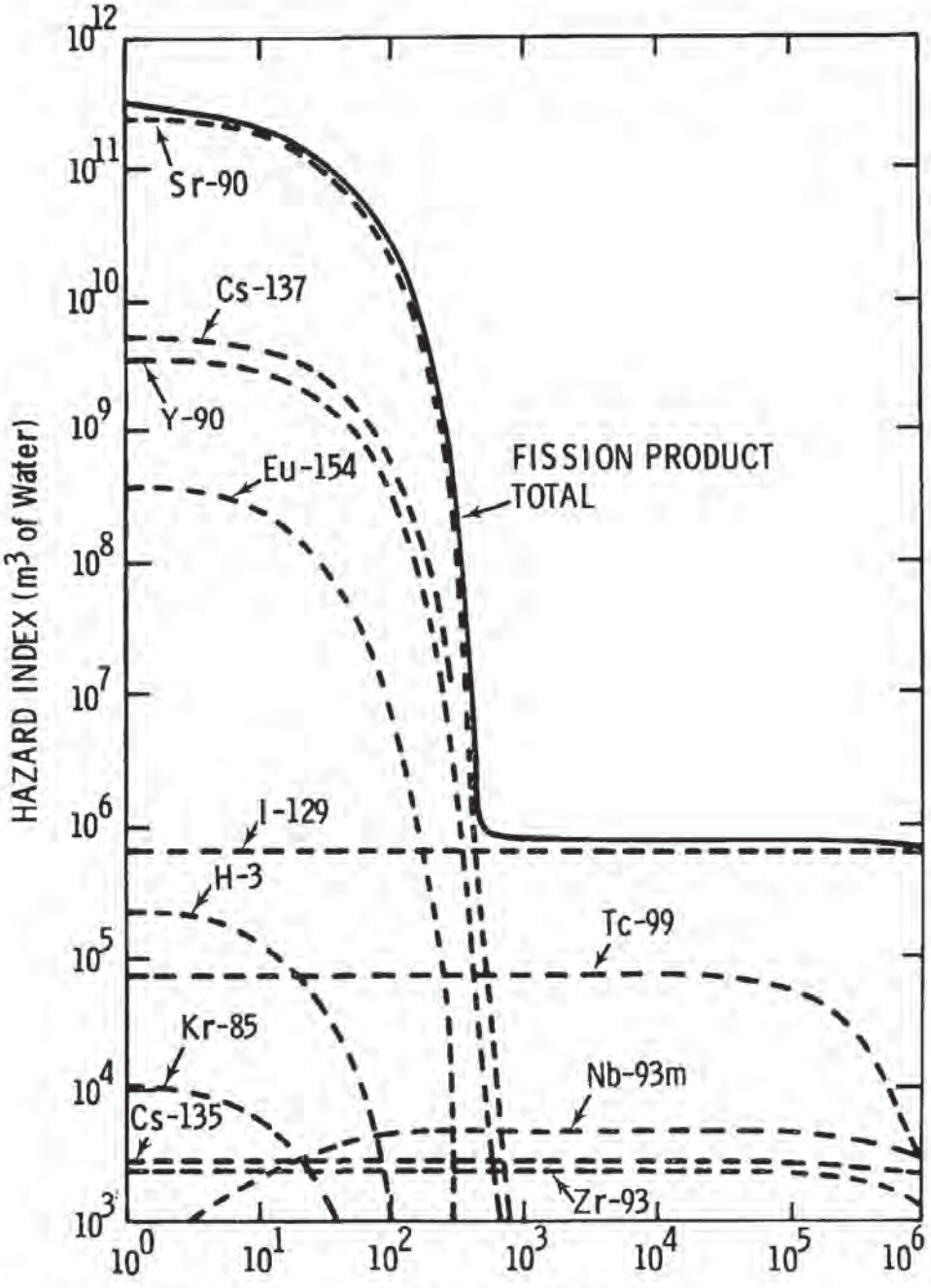


Illustration from Ref. 11

FIGURE 4.9 - FISSION-PRODUCT HAZARDS FROM A 1000-MWE PWR
(WASTES FROM 1 METRIC TON OF FUEL AT 34,000 MW_{TH} D)

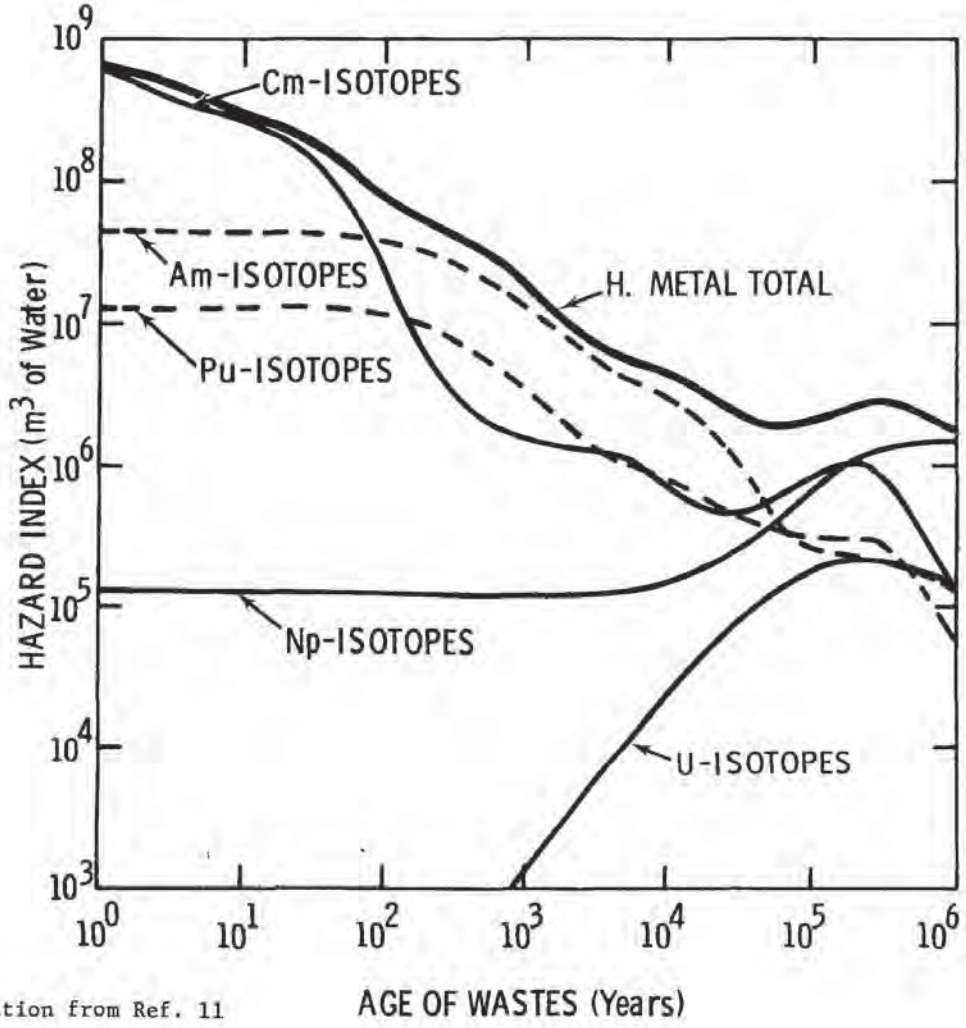


Illustration from Ref. 11

FIGURE 4.10 - HEAVY-METAL ISOTOPE HAZARDS FROM A 1000-MWE PWR (WASTES FROM 1 METRIC TON OF FUEL AT 34,000 MW_{TH} D)

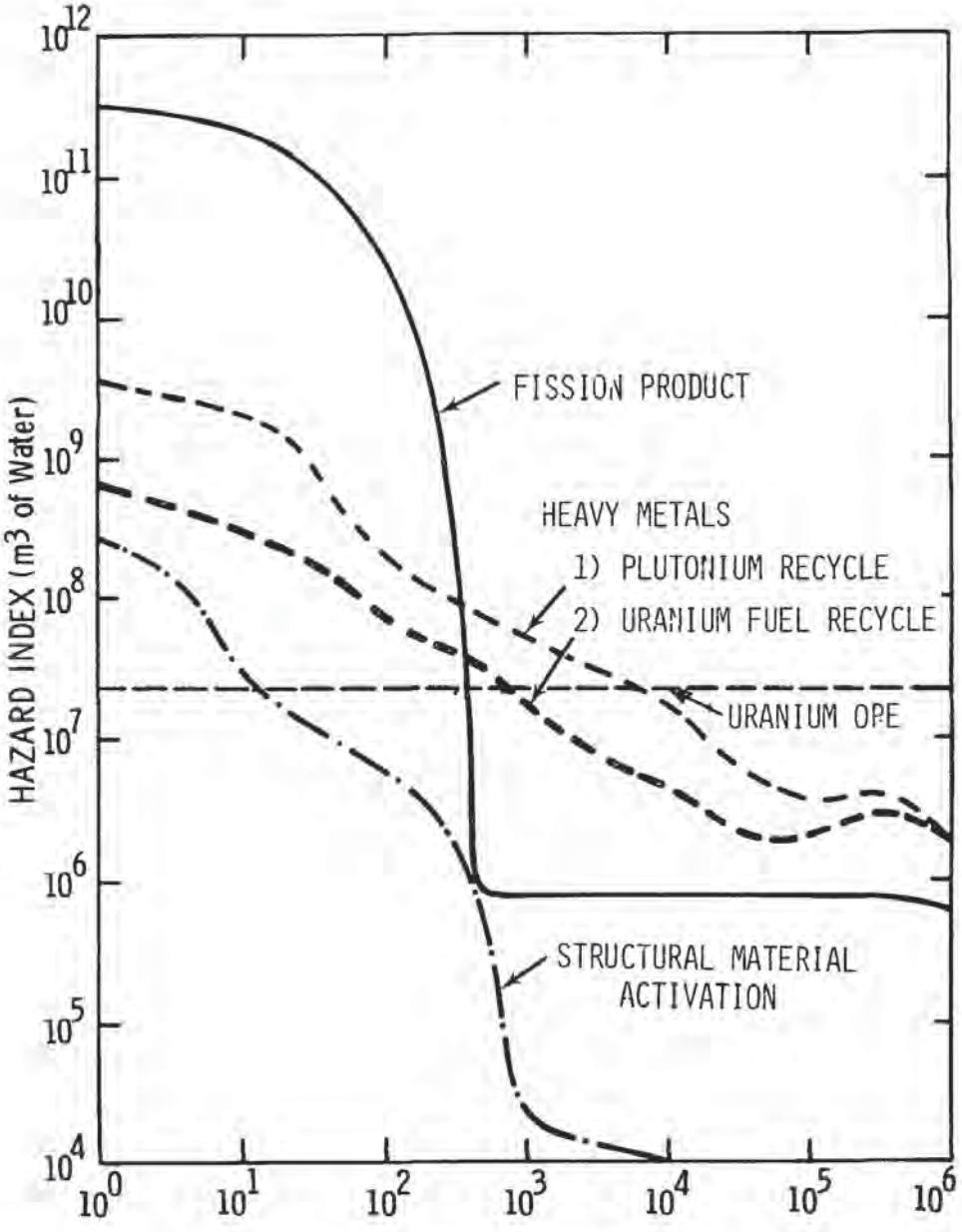


Illustration from Ref. 11

FIGURE 4.11 - SUMMARY OF HAZARD-INDEX COMPONENTS FROM A 1000-MWE PWR (WASTES FROM 1 METRIC TON OF FUEL AT 34,000 MW_{TH} D)

species and to be in sorption equilibrium at all points in the soil column. The biosphere transport model used to describe the impact of surface waste is that discussed in Refs. 18-20. The results of these assessments are given in Tables 4.19 to 4.22 for a leach rate of 0.3% per year, a soil column of 10 miles, and release times of 10^2, 10^4, 10^5, and 10^6 years after closure of the repository.

The doses are calculated for a "maximum" individual whose dietary and living habits, as well as his proximity to the discharge from the soil column, maximizes these doses. In calculating the dose for each discharge peak, the radionuclides are assumed to accumulate in the biosphere for 50 years at the peak discharge rate. The individual is then assumed to be exposed to the accumulated radionuclides for 50 years following that initial 50-year buildup period.

Results given in Ref. 2 are difficult to generalize because of their scope and complexity. The executive summary of that document describes them as follows:

> "The results of this study showed that for reasonable storage conditions the potential incremental radiation doses would be of the same order as, or less than, doses from natural sources... The study required numerous assumptions concerned with the transport of radioactivity from the geologic storage site to man. The assumptions used, on the whole, maximized the estimated potential radiation doses...
> The study found that there are feasible conditions of geologic emplacement where the predicted incremental dose to man is calculated to be as low as one tenth of background...."

Subsequent studies have extended the transport models to media other than "western desert soil," and the results have been extended to wastes other than high-level.[21-27]

An analysis of the consequences of meteorite impact appears in Ref. 1. The repository was assumed to contain all HLW generated by the commercial U.S. nuclear power industry through the year 2000.

The consequences of the impact of a meteorite sufficiently large to breach the repository would be twofold. The first would be due to waste ejected into the upper atmosphere so that it was dispersed globally. The consequences of such dispersion were found to be minor. The second is due to contamination of the area surrounding the repository. The potential consequences for this are given in Tables 4.2.3 and 4.2.4. The total area contaminated is roughly 10^3 square kilometers. Best estimates of the probability of these consequences is of the order of 10^{-13} per year[1] (once in 10 trillion years) and the mechanical damage of the impact would be extremely severe, independent of radiological effects.

Modeling studies such as outlined above take into account the dynamics of repository failure and thereby overcome limitations associated with the RHI concept. The validity of the results of such studies depends strongly, however, on the validity of the modeling assumptions and the accuracy of the data that are the bases for numerical values used in the models.

The data of most importance are those pertaining to the structure and properties of the repository geology and those pertaining to nuclide interactions with the geosphere and biosphere pathways. Both types of data are site-specific. Available information on pathway interactions is extensive[28] and representative of the types of media expected to be used for repositories. Thus, there is reason to believe that results of past studies, which have been generic, are representative of results that will be obtained for specific sites. There are ongoing programs to extend the data base and to evaluate possibilities that interaction phenomena change with time as a result of changes in physical and chemical properties of pathway materials and radioactive nuclides.

Status of Long-Term Risk Assessments

Although results of long-term risk assessments for waste repositories cannot be verified by experiments or experience, there are methods for making such assessments, and results from past studies indicate the risks can be expected to be small.

Past work does not, however, fully cover the risk situations that should be explored. Representative studies have been done with varying degrees of depth and sophistication, but there are still uncertainties in areas such as the effect of waste presence on repository stability; the probabilities and consequences of intrusive acts by humans; the validity of data used in modeling studies; the design and regulatory actions needed to minimize possibilities of repository failure; projection of future societal habits and demography, and, finally, the relative importance of various potential initiating events.

These uncertainties have permitted considerable controversy concerning the long-term risks and impacts of disposal of radioactive wastes. Some claim that the risks from radioactive waste disposal are unacceptable or that no general solution for disposal has been adequately proven.[29-31] For example, Hollocher has stated that the proposed plans are either

> "...dubious in concept (caverns excavated by nuclear weapons, depositing wastes in liquid form in rock caverns), not technically feasible (disposal in solar orbit), or they are so dependent upon site-specific geological characteristics that suitability cannot be determined a priori without extensive on-site investigation (disposal in bedded salt or under Antarctic ice)."[30]

Others feel confident that risks will be acceptable and that safe disposal is feasible though not yet implemented.[32-34] For example, Cohen favorably compares the long-term risks of ingestion of HLW with ingestion hazards associated with other materials in the U.S. soil such as uranium ore, copper, chromium, nickel, aluminum, and biological agents.[33]

Results of these assessments are not subject to proof in the usual demonstration sense. Their evaluation will depend on one's degree of confidence in the validity of the results and on subjective judgments as to what levels of risk are acceptable.

Past studies were made for conditions similar to those expected for actual repositories. Studies of waste transport from repositories by water movement indicate that potential radiation doses to man would be of the order of background. However, site-specific studies, using actual site data, will be needed to provide estimates of risks that might result from actual implementation.

Table 4.19

Summary of 50-Year Accumulated Dose to Maximum Individual
Time of Initial Release After Yr-2000: 100 Yr
Leach Rate: 0.3%/Yr Path Length: 10 Mile

Years Since Burial in Year 2000*	Nuclides Contributing to Dose	Dose, mrem		
		Whole Body**	GI-LLI	Bone
2.50E2	^{99}Tc, ^{129}I	2.3E2	2.6E4	5.4E2
1.55E2	^{14}C	6.3E4	3.8E4	3.1E5
3.73E3	^{93}Mo	3.5E-1	2.1E0	2.5E-7
1.46E4	^{41}Ca, ^{79}Se, ^{237}Np	1.3E3	2.0E3	1.2E4
4.85E4	^{59}Ni	4.5E1	2.0E1	2.7E2
7.26E4	^{87}Rb, ^{225}Ra, ^{229}Th, ^{233}U, ^{237}Np	9.1E-2	7.1E-2	5.7E-1
1.45E5	^{135}Cs	3.5E2	2.0E1	9.4E2
1.67E5	^{107}Pd, ^{126}Sn	8.2E2	2.4E3	4.6E3
3.6E5	^{225}Ra, ^{229}Th, ^{233}U	2.1E-2	2.7E-2	2.3E-1
4.82E5	^{225}Ra, ^{229}Th, ^{233}U, ^{247}Cm	1.3E-2	1.7E-2	1.4E-1
1.45E6	^{93m}Nb, ^{210}Pb, ^{210}Po, ^{226}Ra, ^{242}Pu	7.5E-1	6.5E2	7.5E0
2.07E6	^{210}Pb, ^{210}Po, ^{226}Ra, ^{236}U	1.1E3	2.9E1	2.4E3
2.41E6	^{210}Pb, ^{210}Po, ^{226}Ra	3.4E1	7.3E-1	6.8E1
7.23E6	^{224}Ra, ^{228}Th, ^{232}Th	6.3E-3	3.7E-2	9.6E-2

*Elapsed time between burial and the time at which the concentration in water emerging from the soil column reaches its maximum.

**The 50-year accumulated total body dose to an individual due to natural background is about 5000 mrem.

Table 4.20

Summary of 50-Year Accumulated Dose to Maximum Individual
Time of Initial Release After Yr-2000: 10,000 Yr
Leach Rate: 0.3%/Yr Path Length: 10 Mile

Years Since Burial in Year 2000*	Nuclides Contributing to Dose	Dose, mrem		
		Whole Body**	GI-LLI	Bone
1.0E4	^{99}Tc, ^{129}I	2.2E2	2.5E4	5.2E2
1.1E4	^{14}C	1.9E4	1.1E4	9.4E4
1.4E4	^{93}Mo	1.7E-1	1.0E0	1.2E-7
2.5E4	^{41}Ca, ^{79}Se, ^{237}Np	1.4E3	2.2E3	1.3E4
5.8E4	^{59}Ni	4.2E1	1.8E1	2.5E2
8.2E4	^{225}Ra, ^{229}Th, ^{233}U, ^{237}Np	7.2E-2	4.8E-2	3.6E-1
1.6E5	^{135}Cs	3.3E2	1.9E1	8.7E2
1.7E5	^{107}Pd, ^{126}Sn	7.6E2	2.3E3	4.3E3
4.9E5	^{225}Ra, ^{229}Th, ^{233}U, ^{247}Cm	1.1E-2	1.4E-2	1.2E-1
1.5E6	^{93m}Nb, ^{210}Pb, ^{210}Po, ^{226}Ra, ^{242}Pu	8.6E-1	6.6E2	7.6E0
2.1E6	^{210}Pb, ^{210}Po, ^{226}Ra, ^{236}U	1.7E3	2.9E1	3.4E3
7.2E6	^{224}Ra, ^{228}Th, ^{232}Th	6.2E-3	3.7E-2	9.5E-2

*Elapsed time between burial and the time at which the concentration in water emerging from the soil column reaches its maximum.

**The 50-year accumulated total body dose to an individual due to natural background is about 5000 mrem.

Table 4.21

Summary of 50-Year Accumulated Dose to Maximum Individual
Time of Initial Release After Yr-2000: 100,000 Yr
Leach Rate: 0.3%/Yr Path Length: 10 Mile

Years Since Burial in Year 2000*	Nuclides Contributing to Dose	Dose, mrem		
		Whole Body**	GI-LLI	Bone
1.0E5	^{99}Tc, ^{129}I	1.7E2	1.8E4	3.9E2
1.01E5	^{14}C	3.6E-1	2.1E-1	1.8E0
1.2E5	^{41}Ca, ^{79}Se, ^{237}Np	1.0E3	1.9E3	1.3E4
1.5E5	^{59}Ni	1.9E1	8.5E0	1.1E2
2.4E5	^{135}Cs	3.3E2	1.9E1	8.7E2
2.6E5	^{126}Sn	4.1E2	1.4E3	2.3E3
2.17E6	^{210}Pb, ^{210}Po, ^{226}Ra, ^{236}U	1.4E3	2.6E1	2.9E3

*Elapsed time between burial and the time at which the concentration in water emerging from the soil column reaches its maximum.

**The 50-year accumulated total body dose to an individual due to natural background is about 5000 mrem.

Table 4.22

Summary of 50-Year Accumulated Dose to Maximum Individual
Time of Initial Release After Yr-2000: 1,000,000 Yr
Leach Rate: 0.3%/Yr Path Length: 10 Mile

Years Since Burial in Year 2000*	Nuclides Contributing to Dose	Dose, mrem		
		Whole Body**	GI-LLI	Bone
1.0E6	^{99}Tc, ^{129}I	2.7E1	9.8E2	3.2E1
1.01E6	^{237}Np	5.9E2	1.3E3	9.3E3
1.05E6	^{59}Ni, ^{226}Ra, ^{229}Th, ^{233}U	2.6E-2	2.6E-2	2.2E-1
1.14E6	^{135}Cs	2.7E2	1.6E1	7.2E2
1.16E6	^{126}Sn, ^{226}Ra, ^{229}Th, ^{233}U	5.3E-2	4.3E2	4.8E0
3.07E6	^{210}Pb, ^{210}Po, ^{226}Ra, ^{236}U	5.7E2	1.3E1	9.3E2

*Elapsed time between burial and the time at which the concentration in water emerging from the soil column reaches its maximum.

**The 50-year accumulated total body dose to an individual due to natural background is about 5000 mrem.

Table 4.23

Calculated Population Doses for Meteorite Impact after 1000 Years of Decay[1]

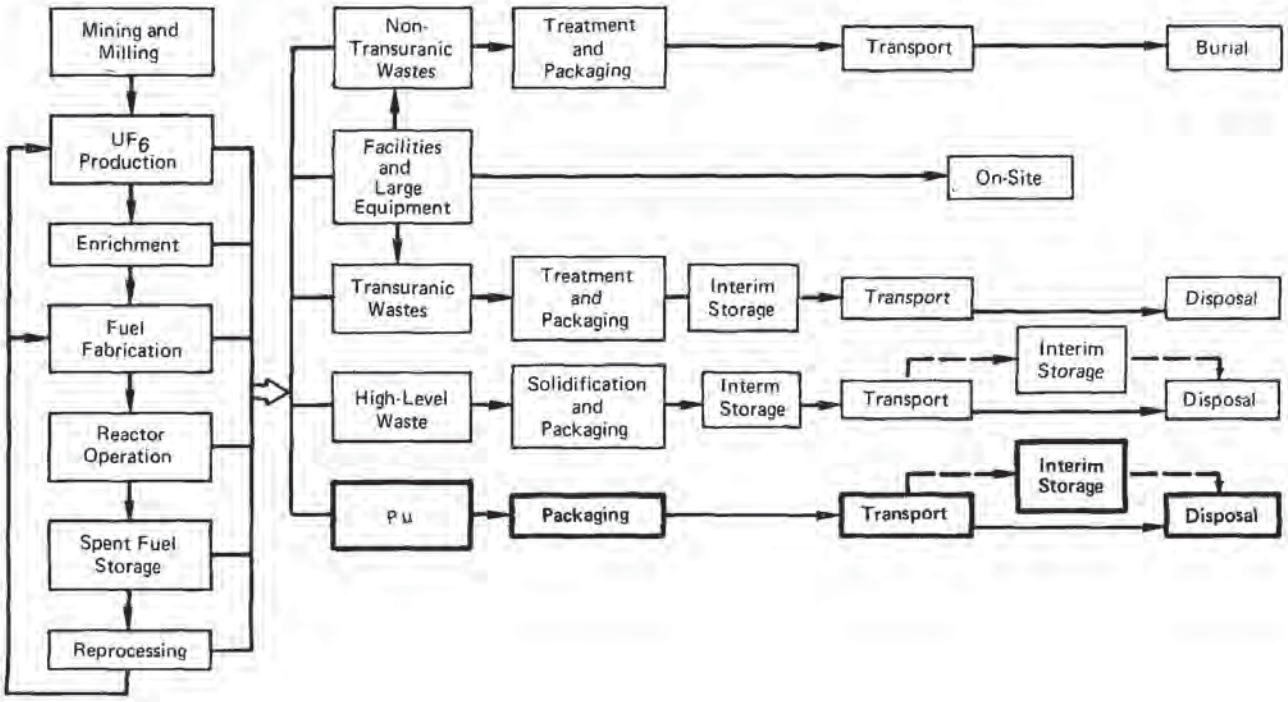
Radio-Nuclide	50-Year Dose Commitment (rem) Due to First Year of Exposure				
	Whole Body Dose from Exposure to Ground Surface	Bone Dose from Inhalation of Resuspended Activity	Organ Dose Due to Ingestion via the Terrestrial Food Chain		
			Whole Body	Bone	G. I. Tract
^{210}Pb	7.3×10^{-6}	4.6×10^{-7}	2.0×10^{-3}	5.6×10^{-2}	3.8×10^{-5}
^{210}Po	0	1.4×10^{-8}	4.5×10^{-3}	1.8×10^{-2}	3.3×10^{-3}
^{226}Ra	1.3×10^{-6}	3.2×10^{-6}	2.8	2.7×10^{1}	5.7×10^{-3}
^{230}Th	1.4×10^{-5}	2.4×10^{-4}	5.0×10^{-4}	1.8×10^{-2}	5.6×10^{-4}
^{234}U	2.1×10^{-3}	1.2×10^{-4}	1.3×10^{-1}	2.1	1.6×10^{-2}
^{237}Np	5.7×10^{-2}	8.5×10^{-2}	4.8×10^{-1}	1.2×10^{1}	5.6×10^{-1}
^{238}Pu	1.1×10^{-2}	1.7×10^{-1}	8.9×10^{-2}	3.5	3.3×10^{-1}
^{239}Pu	7.0×10^{-2}	3.4	1.7	7.0×10^{1}	5.8
^{240}Pu	2.7×10^{-1}	5.7	2.9	1.2×10^{2}	9.8
^{241}Am	1.9×10^{1}	2.4×10^{1}	1.2×10^{2}	1.8×10^{3}	1.4×10^{2}
^{242}Am		6.1×10^{-2}			
^{243}Am	9.7	3.6	1.8×10^{1}	2.7×10^{2}	2.2×10^{1}
^{242}Cm	7.6×10^{-3}	1.2×10^{-3}	5.2×10^{-3}	7.8×10^{-2}	3.7×10^{-1}
^{243}Cm		3.5×10^{-2}			
Total	2.9×10^{1}	3.7×10^{1}	1.5×10^{2}	2.3×10^{3}	1.8×10^{2}

Table 4.24

Calculated Population Doses for Meteorite Impact After 100,000 Years of Decay[1]

Radio-Nuclide	50-Year Dose Commitment (rem) Due to First Year of Exposure				
	Whole Body Dose from Exposure to Ground Surface	Bone Dose from Inhalation of Resuspended Activity	Organ Dose Due to Ingestion via the Terrestrial Food Chain		
			Whole Body	Bone	G. I. Tract
^{210}Pb	2.6×10^{-3}	2.0×10^{-4}	7.4×10^{-1}	2.0×10^{1}	1.4×10^{-2}
^{210}Po	0	5.1×10^{-6}	1.6	6.6	1.2
^{226}Ra	5.0×10^{-4}	1.2×10^{-3}	1.0×10^{3}	9.8×10^{3}	
^{230}Th	9.0×10^{-4}	1.5×10^{-2}	3.1×10^{-2}	1.1	3.6×10^{-2}
^{233}U	2.3×10^{-3}	2.0×10^{-4}	2.2×10^{-1}	3.6	2.7×10^{-1}
^{234}U	9.9×10^{-4}	1.0×10^{-4}	1.0×10^{-1}	1.6	1.3×10^{-1}
^{238}U	5.5×10^{-5}	1.8×10^{-6}	1.9×10^{-3}	3.1×10^{-2}	2.0×10^{-3}
^{237}Np	6.2×10^{-2}	9.3×10^{-2}	5.2×10^{-1}	1.3×10^{1}	6.1×10^{-1}
^{239}Pu	1.1×10^{-2}	5.3×10^{-1}	2.6×10^{-1}	1.1×10^{1}	9.0×10^{-1}
^{242}Pu	5.0×10^{-4}	1.1×10^{-2}	5.6×10^{-3}	2.2×10^{-1}	2.0×10^{-2}
^{243}Am	1.2×10^{-3}	4.0×10^{-4}	2.2×10^{-3}	3.4×10^{-2}	2.7×10^{-3}
Total	8.1×10^{-2}	6.5×10^{-1}	1.0×10^{3}	9.8×10^{3}	1.0×10^{3}[a]

[a] These totals assume organ dose from ^{226}Ra equal to total body dose.



4.5 Plutonium as a Waste

In the uranium-only recycle, plutonium produced in reactors is not recovered and used for its fuel value. It is assumed in this discussion that the plutonium will be handled, treated, and disposed of as a waste; the option of storage for possible future use is not considered.

Past philosophy concerning nuclear power has been based on the belief that the uranium and plutonium in spent fuel would be recovered and recycled. We have therefore not found detailed analyses of the technology and impacts for treatment of plutonium as a waste. This discussion is consequently a qualitative extrapolation of known technology for management of similar (e.g., high-level) wastes. The major factors to consider in the extrapolation are the quantities of plutonium in fuel cycle wastes (increased, in uranium-only recycle, by a factor of approximately 100 over quantities for the uranium-plus-plutonium recycle), and the state of aggregation and purity of the plutonium (which has criticality implications).

4.5.1 Technology Selection

Uranium-only recycle requires operation of a reprocessing plant to recover the uranium from spent fuel. The primary plutonium-bearing waste stream could be produced in two ways: by putting the plutonium in the HLW stream, or by isolating and managing it as a separate waste stream. There are two further options for plutonium separation: to leave it contaminated with some fission products, or to carry the separation through to a high degree of plutonium purity.

There are advantages and disadvantages for each of these options:

- Putting the plutonium into HLW simplifies processing but complicates HLW management because of criticality control requirements. High Pu concentrations in HLW could lead to formation of Pu-rich solids and sludges.

- Leaving the Pu contaminated with fission products makes it less attractive as a target for illicit use but requires remote-handling and quite heavily shielded process equipment.

- Purifying the reactor plutonium maximizes process requirements and results in a product that can be subsequently handled without massive shielding, but the product may be an attractive target for diversion attempts.

Technology exists for implementing any of these options, and variations in the environmental impacts for the reprocessing plant would be small. A choice could therefore be made on the basis of factors such as cited above and subsequent waste management operations (transportation and disposal). Considerations for each of the options are discussed in more detail below.

4.5.2 Treatment and Handling of HLW Containing Significant Amounts of Plutonium

The HLW produced when U and Pu are both recovered would, by process design, contain a residual 0.5% of the Pu originally in the spent fuel. Under the uranium-only recycle option, all of the Pu would be present; i.e., Pu quantities in HLW would be increased by a factor of 200.*

If the plutonium is left with the HLW stream, we assume that the mixed waste is handled the same way as conventional HLW in terms of processing steps involved, but with additional requirements throughout the process for criticality control. This includes temporary tank storage for cooling (see Sec. 4.2.1), solidification (Sec. 4.2.2), shipment to the repository (Sec. 4.9), and disposal there. For normal operation at the reprocessing plant, impacts ascribable to the plutonium conversion facility are eliminated; therefore, analysis in Section 4.2 is conservative for this case. Tank storage of HLW with plutonium in such large concentrations has not been analyzed, nor has the solidification process been studied for this mixed waste. Therefore, while the technologies are the same as those presently in use or envisaged for immediate use, details of design and operation may differ because of specific chemistries of the liquid waste.

This alternative simplifies the extra handling problem but also poses many unaddressed questions. In high-burnup uranium fuel from LWRs there is about 10 kg of plutonium per ton (60% plutonium-239). During fuel dissolution by nitric acid, a neutron-absorbing material (usually gadolinium nitrate), is added to prevent criticality. If this material is not later removed from the HLW stream it goes to the liquid HLW

*The overall increase in Pu quantities sent to disposal is only a factor of 100 because process wastes (MOX plant, etc.) from Pu recycle would also be expected to contain 0.5% of the original Pu inventory.

storage tanks, where it is sufficient to prevent criticality provided the waste constituents remain homogeneous. If there are mechanisms available for the selective precipitation of plutonium it would be possible to achieve criticality in the tanks. This same consideration arises in solidification of liquid HLW if there is waste fractionation.

The extra plutonium will raise the neutron radiation level in the waste due to a 20% increase of alpha activity in the waste (alpha-neutron reactions) and spontaneous fission.

4.5.3 Treatment and Handling of Separated Plutonium as a Waste

Complete separation and purification of Pu would involve operations and impacts such as described in Section 4.2. If the plutonium is a waste, the destination of shipments would be a federal waste repository.

This discussion focuses on the option of partial purification of the plutonium. In one of the steps of the Purex process, the plutonium stream contains a significant amount of fission products. Purification could stop at that stage and the material could be converted to an oxide and dried. The resulting product would be a highly gamma-radioactive dry, fluffy powder with a bulk density of about 2 gm/cm^3. All operations would have to be performed remotely and with shielding.

A previous analysis (GESMO, Vol. 3, pp. IV H-18 and H-19) has suggested that the contaminated plutonium waste product could be packaged in steel cylinders two feet long and four inches in diameter. About 55 such cylinders would be produced per metric ton of fuel reprocessed. These cylinders could then be packaged with cladding hulls, which also require shielding, for shipment to the waste repository. The packaging geometry would have to be designed for criticality control. This concept has merit and would be technically feasible, but logistical and other requirements have not been addressed.

Because plutonium has been viewed as a valuable energy source, the literature contains few analyses of its disposal as a waste. Additional considerations of criticality control, transport, and safeguards must be given. A discussion of these considerations follows.

4.5.4 Transport of Plutonium Waste

If the plutonium is left in the HLW, transportation characteristics and impacts would be as described in Section 4.2.4. Additional analysis of the possibility and consequences of an accident leading to criticality would be necessary for detailed designs.

Risks due to truck accidents with separated plutonium shipments have been analyzed in "The Risk of Transporting Plutonium Oxide and Liquid Plutonium Nitrate by Truck," BNWL-1846, August 1975. The risk calculations were based on truck accident

environment data developed by Sandia Laboratories.* The likelihood that a plutonium truck shipment would be involved in an accident was estimated to be about one in 1-1/2 years. Most accidents would not involve any release of plutonium. For projected shipping rates in the early 1980's, the likelihood of a release was estimated to be once in 330 years for oxide powder in the conventional shipping container.

4.5.5 Disposal of Plutonium Waste

Since PuO_2 has been perceived to date as an energy resource, analyses oriented to PuO_2 as a waste for disposal do not exist in the literature. For this discussion, we assume a bedded-salt repository as described in Section 4.4, and a waste form of either cylindrical capsules containing solid PuO (as described in GESMO, Vol. 3, pp. IV H-18 and H-19), or Pu solidified with the HLW in glass. The effects of either option on bedded-salt disposal will be similar: the long-term hazard index (after about 600 years) of the repository inventory will be significantly increased (by a factor of about 100) owing to the increased amount of Pu relative to other waste constituents. This increase in the hazard index will primarily affect long-term risk calculations (Sec. 4.4.2).

The effects of large amounts of Pu on repository size have not been analyzed in the literature, but a conservative approximation can be made with the following analysis. Using Cheerton and Turner's results for an alpha mine (Sec. 4.2.6), scaling the thermal response (ΔT_{max} = 22°C @ 1000 years with .64 MW of heat) for 10-year-old alpha waste in 180 acres to a ΔT max of 200°C, using 117 W (of Pu)/MTHM (from GESMO, Vol. 3, p. IV H-39), and using 35-MTHM/RRY and a 4000 RRY/repository, we can calculate the underground area necessary to limit the temperature excursion to 200°C. This calculation indicates an area of 500 acres. Since the total underground acreage for 4000 RRY was for the baseline case (2000 acres), the underground workings would, on this conservative basis, be $\sim$ 25% larger. If the repository underground area were changed from 2000 acres to 2500 acres, and the buffer radii were kept constant, the percentage change in land commitment would be much smaller. Hence the 25% increase figure is considered conservative. An analysis given in GESMO (Table IV H-24), indicates the repository area impact of plutonium waste would be about 20%.

Releases to the environment during repository operations are discussed in Section 4.4, which indicates, in Table 4.17, that release of transuranics would be 1.1×10^{-5} Ci/RRY. As a first approximation, disposal of plutonium in the repository would increase the release of transuranics by a multiple equal to the increase in plutonium inventory in the repository; i.e., a factor of 100. TRU release at the repository would then be 1.1×10^{-3} Ci/RRY (see Sec. 4.4.1).

Another factor to consider for plutonium waste in a repository is the possibility of a criticality accident. Emplacement procedures can assure that such an accident

*J. T. Foley, W. F. Hartman, D. W. Larson, and R. K. Clarke, "Quantitative Characterization of the Environment Experienced by Cargo in Motor Carrier Accidents," Proceedings of the 4th International Symposium on Packaging and Transportation of Radioactive Materials, Miami Beach, Florida, September 22-27, 1974.

cannot occur during operations or, as a result of emplacement configuration, in the long term if the emplacement configuration is maintained. The long-range uncertainty, which has not been investigated to date, is the possibility that geologic movements or other long-term natural processes could cause a critical configuration to form. If criticality was approached, however, the resulting energy releases would tend to prevent rapid (bomb-like) assembly.

Estimates of conditions required to produce criticality ($k_{eff} \geq 1.0$) in a salt repository are shown in Figures 4.12 and 4.13 for slab and spherical geometries respectively. Assuming distant spacing during emplacement (e.g., canisters separated by approximately 5 to 10 meters, as for HLW), massive geologic movements or extensive dissolution of the salt would be necessary to produce critical configurations. As described in Section 4.4.2, such events are considered highly unlikely even in the long detoxification periods involved. If they should occur, the dislocations might extend to the surface, with extensive disruption of geology and the environment. Alternatively, if the disruption were confined to depths such as those at the repository, criticality phenomena would not be noticed at the surface.

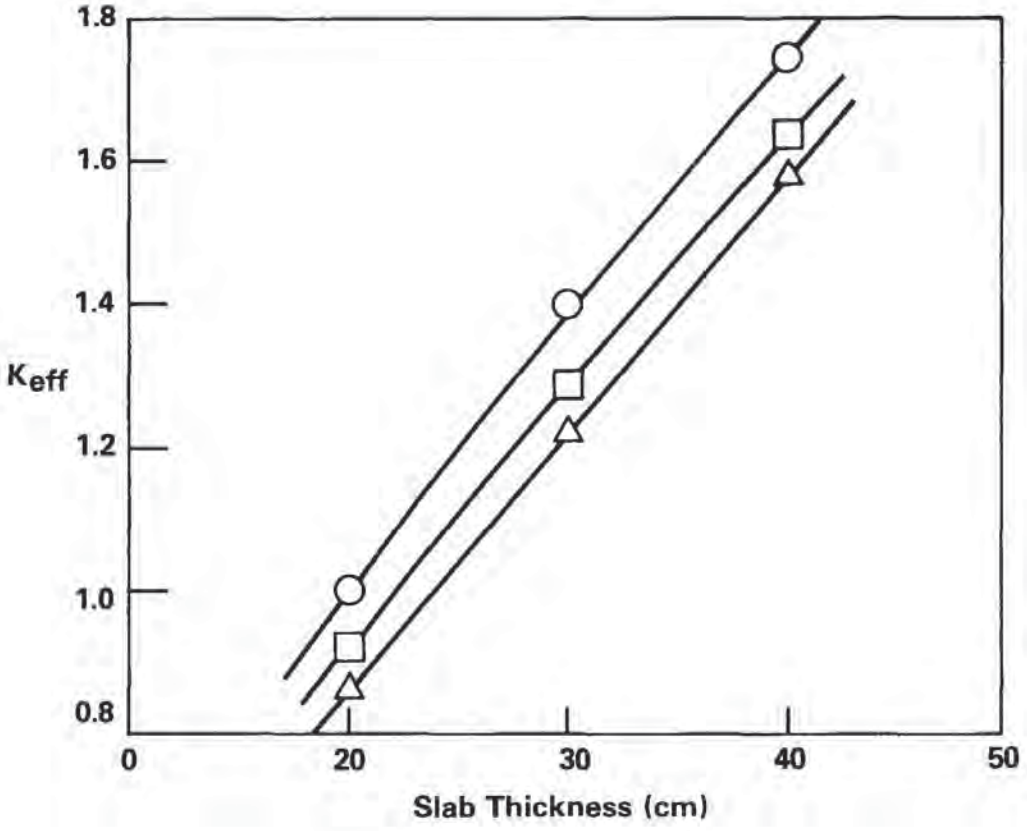


FIGURE 4.12 - (K_{EFF} FOR PLUTONIUM WASTE IN A SALT REPOSITORY (SLAB GEOMETRY)

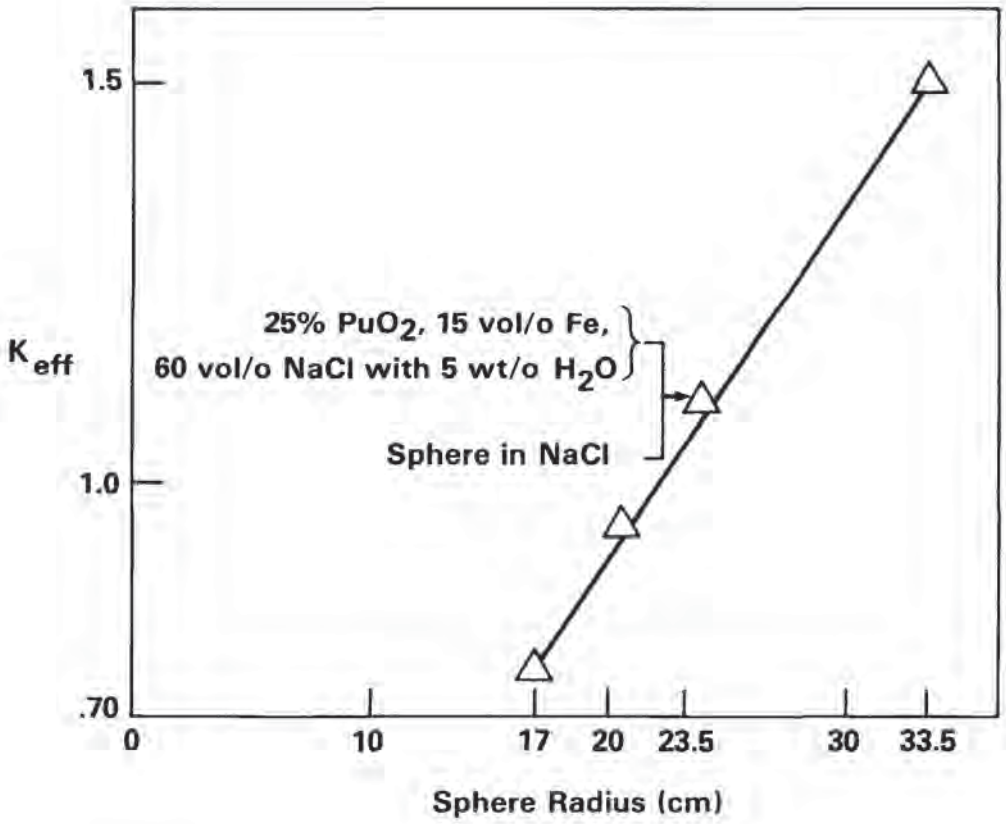
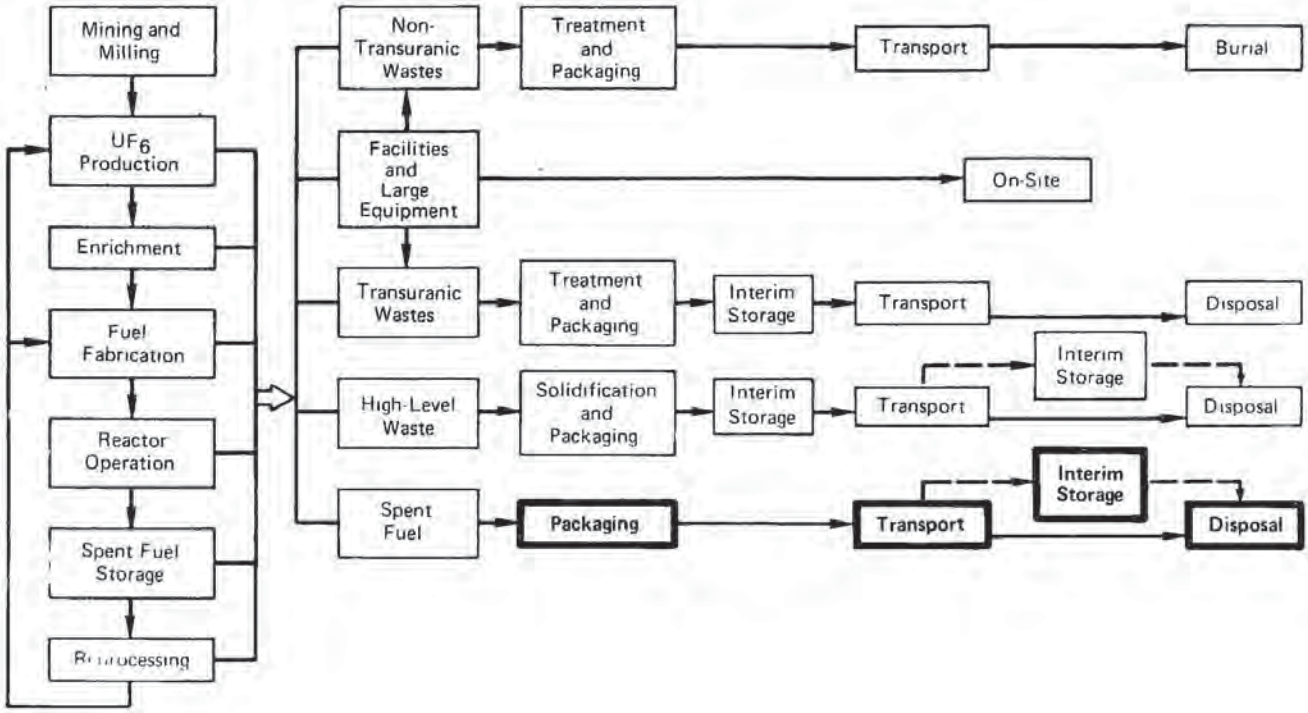


FIGURE 4.13 - (K_{EFF} FOR PLUTONIUM WASTE IN A SALT REPOSITORY (SPHERE GEOMETRY)



4.6 Spent Fuel as a Waste

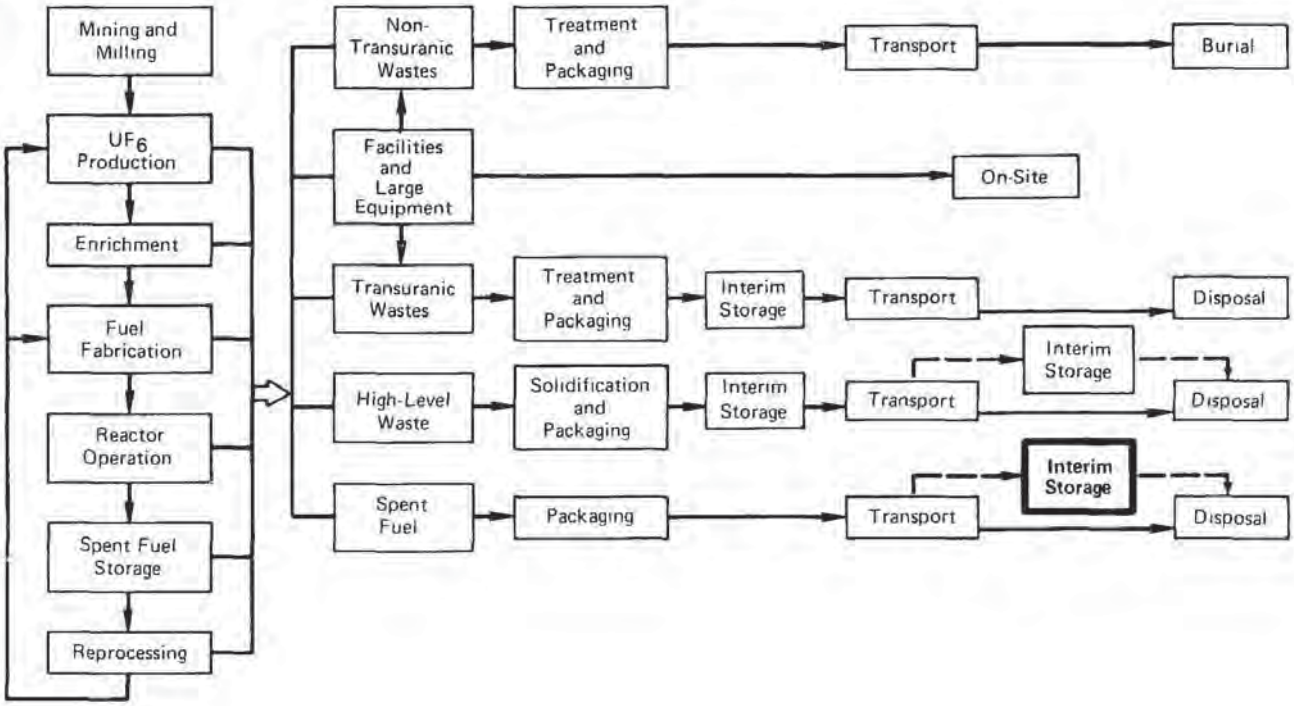
While there are changes in the balance of kinds of waste and the concomitant impacts of waste management between the throwaway fuel cycle and either of the other options, there are only three operations in the waste management sequence when spent fuel is considered to be waste. These are: (1) extra storage of spent fuel, (2) special fuel packaging for disposal, and (3) disposal of the spent fuel as a waste.

The impact of spent-fuel disposal has been evaluated for this Supplement on the basis of retaining the fuel assemblies in water basins for 10 years before packaging, followed by disposal in bedded salt. The fuel packages are assumed to provide containment for the fission products during active management of the fuel as a waste. After geologic disposal, all fission-product gases are assumed to be released from the repository by the failure of the package and fuel cladding, while the solid fission products are retained by the salt.

Discharged fuel is stored mostly in water basins at the reactor site, but some may be shipped to basins at other sites for interim storage. It is assumed that about 10 years after reactor discharge, the fuel will be enclosed in canisters which would then be transferred to a federal repository for disposal. In some cases, two shipments may be necessary--from reactor basin to interim basin, and from basin to repository. The assemblies could be packaged either at the basin storage location or at the repository. The following sections discuss each of these management processes.

Spent-fuel disposal decreases all environmental impacts of the fuel cycle attributable to reprocessing and post-fission waste management because the fuel reprocessing, MOX fuel fabrication, and plutonium storage steps are eliminated.* The major wastes generated are non-TRU low-level wastes from fuel storage, packaging, and transportation, in addition to the spent fuel itself.

*Note: See Ref. 4 for assessment of differences across the entire fuel cycle. Impact at mines, mills, and enrichment processes increases with the throwaway cycle.



4.6.1 Spent-Fuel Storage

Additional storage of spent fuel will be required in the throwaway cycle (as compared to uranium recycle, where reprocessing takes place). This is the subject of a generic environmental impact statement now being prepared by NRC[1], but considerable experience has accrued over the past decades with the storage of spent fuel at reactor sites, and from this experience some assessment of the impacts of that procedure can be made.

4.6.1.1 Technology Selection

Spent-fuel storage underwater in stainless-steel-lined concrete basins is proven technology[2] and is identical to that used in the other fuel cycle options. It is therefore the technology chosen for this analysis. Options have been proposed by other countries[3] but experience with those options is limited to tests.

4.6.1.2 Process Description of Water Basin Storage

It is predicated that spent fuel will be stored for 10 years to reduce radiation and heat emissions at the time of packaging and disposal. When fuel is recycled, the storage time before reprocessing is taken to be one year; thus more assemblies are in water basin storage at any given time with the no-recycle mode.[5] The large number of assemblies to be stored favors close spacing in storage, which can be achieved by the use of neutron-absorbing partitions in the storage racks.

4.6.1.3 Environmental-Impact Bases for Water Basin Storage

It is estimated that for the projected growth of the nuclear industry (Section 3.0) the number of spent fuel assemblies in storage or ultimate disposal by the year

2000 would increase from 37,000 if spent fuel were recycled, to 400,000 with spent-fuel disposal.[6]

The quantity of fuel in basin storage depends on the relative rates of discharge from reactors and removal from basins to geological disposal. The estimate of rates in Ref. 6 indicates that the quantity of spent fuel in basin storage in 1975 would be the same for any fuel cycle option but would be over 4 times greater in the year 2000 with spent-fuel disposal than with uranium recycle. The environmental impact of water basin storage is based on this larger storage, integrated over the years 1975 to 2000.

4.6.1.4 Environmental Impact of Water-Basin Storage

Table 4.25 lists some of the 26-year integrated environmental impacts of extended spent-fuel storage in the throwaway option.[7] The heat dissipation, krypton-85 releases, and solid-waste volume generated by spent fuel in storage over a given time are larger for the throwaway option than for the recycle options because more fuel is stored. The krypton-85 results from the assumed leakage of a small fraction (~1%) of the total krypton-85 contained in the fuel. Environmental impacts other than those listed are similar to those for the recycle options.

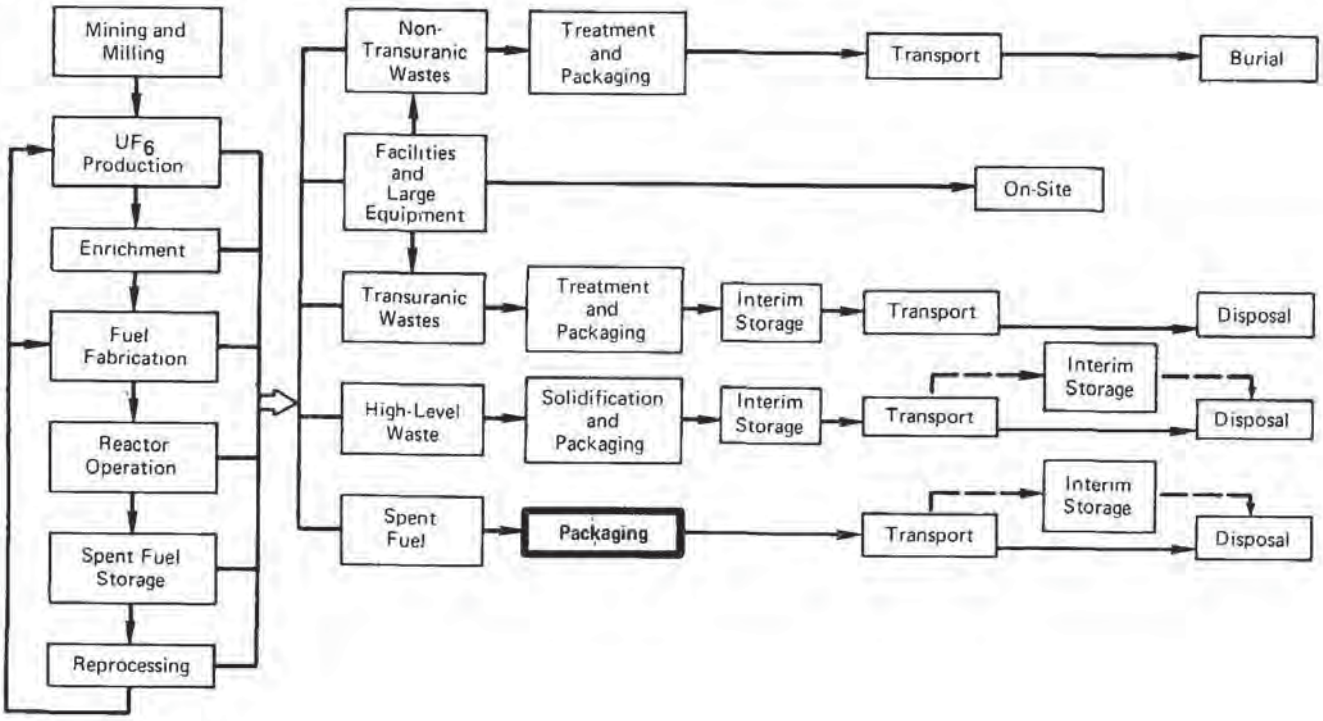
Table 4.25

IMPACT OF EXTENDED WATER BASIN STORAGE (NORMAL OPERATION)

	Cumulative, 1975 to 2000[c]	Normalized to Model Reactor Year[d]
Heat dissipation (Btu)	400×10^{12}	2.6×10^{7}
^{85}Kr released (curies)	6×10^{5}	350
^{85}Kr dose (person-rem, worldwide)	110	1×10^{-2}
Solid wastes (ft^3)[a]	1.2×10^{5}	70
Occupational dose (person-rem)[b]	1.2×10^{4}	7

Note: a. Average of 1 curie per cubic foot (200 cu ft per 1,000 Mt/yr)
b. 20 person-rem per 1,000 Mt/yr
c. Ref. 7
d. Based on 35 MTHM for 10 years.

NRC staff analysis of modifications to existing reactors indicates that extended spent-fuel storage in appropriate facilities poses no new potential accident conditions.[8,9]



4.6.2 Packaging of Spent Fuel

4.6.2.1 Technology

Fuel assemblies are assumed to be packaged before disposal for increased handling safety and assurance of fission product containment before the repository is sealed. Individual assemblies packaged in steel canisters may be backfilled with helium to provide an inert atmosphere around the fuel and to aid in removing heat from the assembly.*

Packaged zirconium-clad UO_2 fuel assemblies have been stored for 10 years or longer under water at the Savannah River Plant.[10] Also, spent fuel from gas-cooled reactors has been packaged at the Idaho Chemical Processing Plant,[11] and CANDU fuel has been packaged and stored.[3] Portions of this packaging technology are applicable to the packaging of LWR spent fuel.

4.6.2.2 Process Description

Packaging of spent fuel begins with placing the (assumed) 10-year-old fuel in insulated drying wells within a shielded facility to remove residual basin water. After drying, the assembly is placed in a square canister that has a 0.50-inch wall and is slightly larger than the fuel assemblies. A canister for one BWR assembly would weigh 0.35 MT, and for one PWR assembly, 0.48 MT. The larger package would occupy about 8 ft^3. Air is evacuated, helium is charged, and a leak test is made before and after the canister is sealed. This canister and the fuel cladding together provide two containment barriers.

*This package can be shown to maintain the temperature of 10-year-old fuel below 500°C during in-air handling or salt emplacement, even with loss of He. Fuel frequently withstands temperatures up to 600°C in reactors.[12]

It has been estimated[7] that the cladding on 1% of the fuel rods may be breached by the time of packaging. The provision of a second, larger canister (overpack) for fuel assemblies that contain detected failures restores two barriers to those assemblies.

4.6.2.3 Environmental Impact and Bases for Fuel Packaging

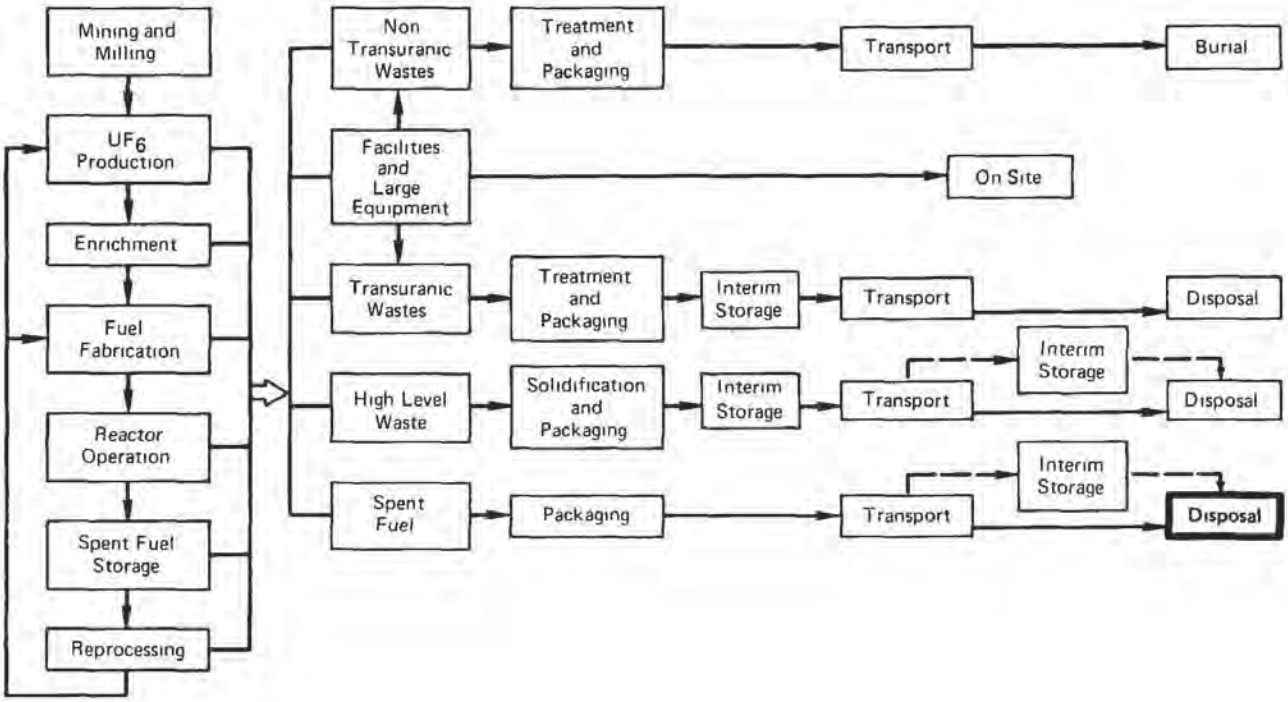
The packaging of fuel assemblies is similar to loading and unloading small spent-fuel casks. The number of assemblies that might need to be packaged between 1986 and the year 2000 is shown in Table 4.26.[5,6] If the assemblies are packaged at the reactor, the environmental impact would be similar to (and would already be included in) the impact of loading and transporting fuel. If the fuel elements are packaged at other locations, up to four plants of 2000-MT/yr capacity would be required. The impact of packaging has not been specifically analyzed. Therefore it is belived that packaging alone doubles the radiological impact of the increased fuel storage that results from the throwaway fuel option.

Table 4.26

NUMBERS OF SPENT-FUEL ASSEMBLIES

Year	Spent Fuel Packaged (MT) Annual	Cumulative
1986	0	0
1987	3500	3500
1988	3500	7000
1989	3500	10500
1990	3500	14000
1991	3570	17600
1992	4100	21700
1993	4600	26300
1994	4900	31200
1995	5500	36700
1996	6100	42800
1997	6700	49500
1998	7200	56600
1999	7800	64400
2000	8400	72800

A packaging facility would generate 1.0 billion Btu/yr of heat per reference reactor, assuming a 6-month inventory of 5-year-old spent fuel generating 2090 watts/MTHM. This value is conservative in that packaging of spent fuel would be more likely to occur after 10 years; therefore the heat generated would be appreciably lower (~50%).



4.6.3 Salt-Bed Disposal of Packaged Spent Fuel

4.6.3.1 Technology

It has been assumed in the past that the uranium and plutonium in spent fuel would be recovered and recycled. Therefore, detailed analyses of the technology of spent-fuel disposal are not to be found in the literature.

The technology for geologic storage is described in Section 4.4, in which salt beds are selected for disposal of HLW canisters. Such repositories could be modified for fuel canister disposal. The technological requirements of repository disposal for all fuel options are approximately the same, although 10 times as many canisters would have to be handled in the spent-fuel throwaway option.

Several spent-fuel assemblies from the Engineering Test Reactor, packaged in stainless steel, were stored for a short period in salt beds during Project Salt Vault at Lyons, Kansas.[13] Data obtained included recommended thermal loading limits in salt, and corrosion attack rates on materials. Successful emplacement techniques for packaged fuel were developed.[13]

4.6.3.2 Description of Fuel Disposal

Operation of repository facilities for the throwaway option could be very similar to that for HLW canisters (Sec. 4.4). Individual canisters containing spent-fuel assemblies would be handled in air, lowered to the mine level in a vertical shaft, then transferred and buried in the salt bed using a shielded transporter.

4.6.3.3 Environmental Impact Bases for Fuel Disposal

A heat load of 150 kW/acre (37 watts/m^2) has been proposed for the salt disposal design.[14] Spacing of containers in the repository is assumed to be governed solely by heat loading, resulting in a spent-fuel spacing of about 5 m for 5-year-old PWR fuel.

The disposal of canisters of spent fuel differs from the disposal of HLW canisters in that:

- There would be about 11 times as many canisters of spent fuel (400,000 spent-fuel canisters v. 37,000 HLW canisters).[6]

- There would be gaseous fission products in spent fuel. Without reprocessing, these gases are retained in the fuel rods. With spent-fuel disposal the gases are assumed to be released from the repository shortly after disposal.

- Plutonium is stored within the fuel assemblies rather than being stored in separate containers in the repository (see Sec. 4.5).

- Fission products present in spent fuel are not in glass form, so long-term releases may be higher if fuel canisters and repository both fail.

With both uranium recycle and spent-fuel disposal, the salt is assumed to retain the solid radioactive fission products. The validity of this assumption has not been evaluated for spent fuel.

4.6.3.4 Environmental Impact of Spent-Fuel Disposal

Environmental impacts from geological storage or disposal of spent fuel are expected to be similar to those from HLW disposal except for radiological impacts. The radiological impact of disposal is increased over that for uranium recycle due to the conservatively assumed release of all fission-product gases present in the fuel (Table 4.27.) These releases are several orders of magnitude greater than those arising from HLW disposal in the repository (Sec. 4.4).

Population doses from these releases (Table 4.28) were calculated by scaling the doses for similar releases at an FRP[15] by the magnitude of associated releases for each isotope (Table 4.27).

Table 4.27

FISSION-PRODUCT GASES RELEASED FROM GEOLOGICAL STORAGE OF SPENT UO_2 FUEL (NORMAL OPERATION)

Gas	Quantity Released Ci/MTU[a]	Annual Release, Ci	
		Per Model Repository[b]	Per Model Reactor[c]
^{3}H	403	2,700,000	14,000
^{14}C	.554	3,700	19
^{85}Kr	8,210	55,400,000	290,000
^{129}I	.0374	252	1.3

[a]Ref. 16 (5-year-old fuel).

[b]Annual addition of 6,750 MTU (15,000 fuel elements) to storage.

[c]For 35 MT annual fuel discharge.

Table 4.28

POPULATION DOSE FROM GEOLOGICAL STORAGE OF SPENT UO_2 FUEL (NORMAL OPERATION)

Organ	Annual Population Dose Normalized To the Model Reactor, Person-Rem
Total Body	260
Bone	620
Thyroid	2,300
Lung	280
Skin	1,500

Disposal of packages containing 112 assemblies per model reactor year, each with a volume of $\sim$ 8 ft^3, would generate a total volume of $\sim$ 875 ft^3 of waste (in square canisters) which may be classified as HLW. For round canisters the comparable value would be $\sim$ 1,200 ft^3.

The most severe accident postulated and analyzed for a geological repository is the rupture of a canister before final placement in the vault.[17] Even with the rupture of a spent-fuel container and the cladding on some fuel rods, dispersal of solids from the fuel would not be significantly (if any) greater than from the calcined wastes used as a basis for evaluating uranium recycle. Although fission-product gases would be released, it is already assumed that they will be released later during storage, and this release is considered in the calculated population dose impact. An increased occupational dose would probably result from such an accident if it occurred in some phases of the disposal operation. This risk would be the same as in fuel transport operations in all fuel-cycle options.

The likelihood of a criticality incident with spent fuel after placement in a salt formation and during the active life of the repository is extremely remote

because even if water were present, canister spacings for heat removal (about 5 m per Sec. 4.6.3.3) are about 10 times those (0.5 m) necessary to prevent criticality.[18] The risk of criticality over geological times because of reassembly of the fissionable components of spent fuel has not been evaluated. Long-term considerations for spent-fuel disposal would be similar to those for plutonium disposal (see Sec. 4.5).

4.6.4 Waste Management Activities Eliminated or Altered With Spent-Fuel Disposal

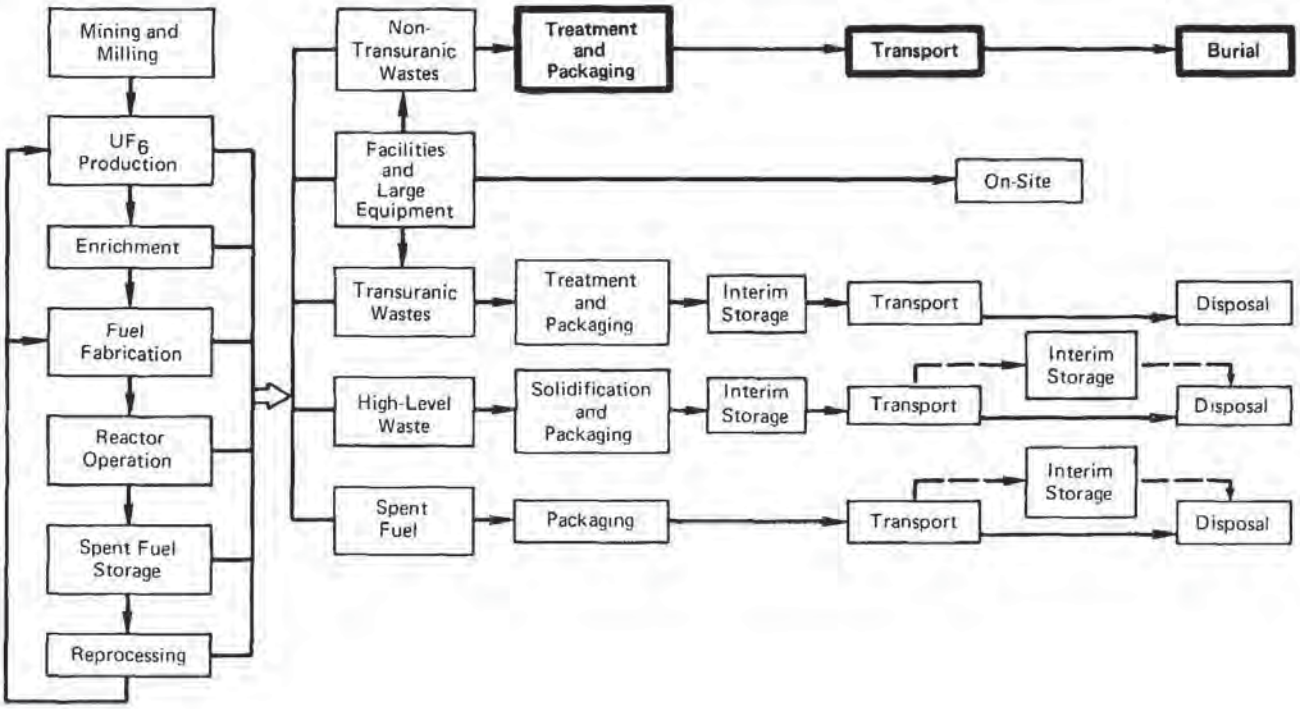
Disposal of spent fuel eliminates several sources of waste and their impacts. The eliminated processes are reprocessing of spent fuel, MOX fabrication, and plutonium storage.

Without reprocessing, the spent-fuel elements themselves constitute the HLW; more than 99% of activity in the waste from the entire cycle would remain in the spent-fuel assemblies.

Also eliminated is most of the impact of processing TRU wastes. Non-TRU wastes increase <0.01% (or by ~ 100,000 gal) if fuel is not recycled.[19]

Steps eliminated by fuel disposal (reprocessing and MOX fuel fabrication) both have low radiological impact per accident and low accident expectancy. Their elimination, however, tends to decrease overall accident risk. The population (environmental) dose commitment due to reprocessing and waste management over the 1975-2000 period will decrease because of the steps eliminated (assuming continued geologic containment).

The projected aggregate of all exposures (occupational plus general population) for the LWR and its fuel cycle industry at the growth projected from 1975 through 2000 is summarized in GESMO.[20]



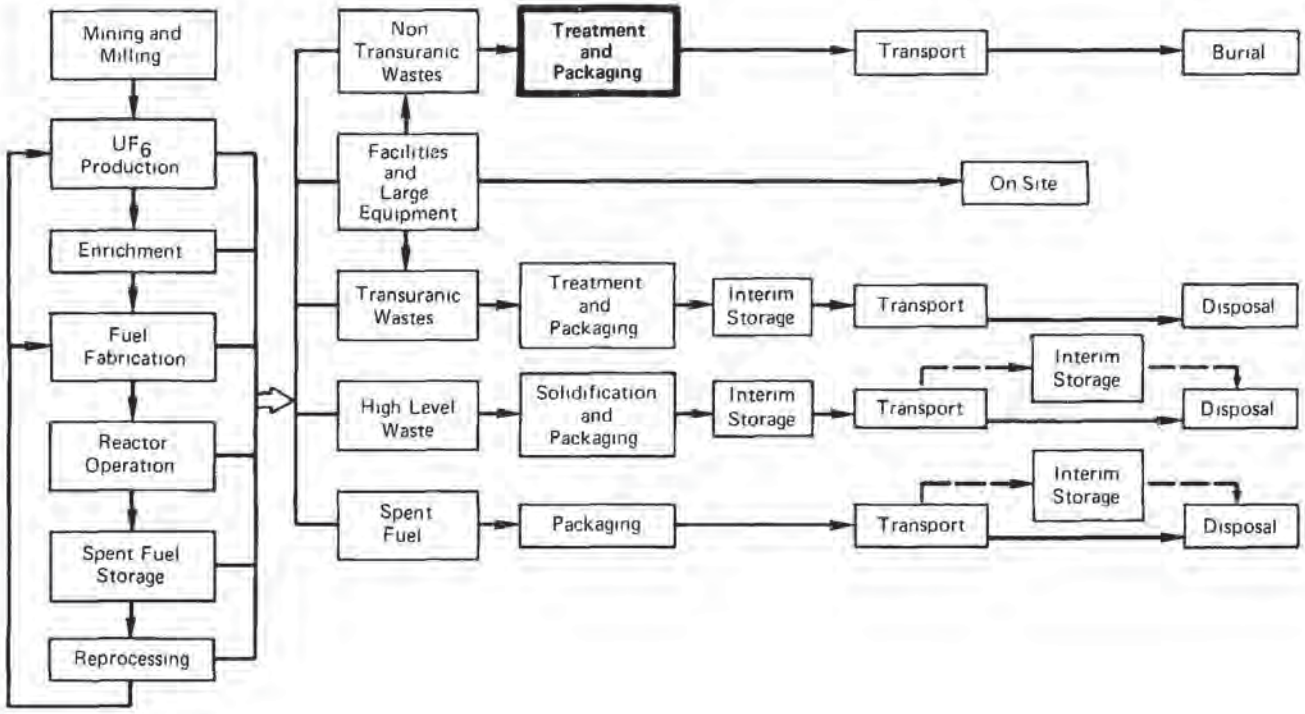
4.7 Management of Low-Level Wastes

The term "low-level" refers to all solid radioactive wastes other than high-level and TRU wastes. These constitute the majority of wastes generated by the LWR fuel cycle (Table 4.29), but which contain a relatively small portion of the radioactivity. The principal radionuclides contained in this waste are cobalt-60, cesium-134, cesium-137, and other lower-yield fission and activation products.[1] These isotopes, with maximum half-lives of approximately 30 years, decay to innocuous levels in tens to hundreds of years, but require isolation during that time.

Table 4.29
LOW-LEVEL WASTE BURIED PER RRY (NORMAL OPERATION)

	UF_6	U Enrich	Fuel Fab.	LWR	Reproc.**	Spent Fuel Storage	Decomm.
Volume (M^3)	53	30	28	630	7	.21	360
Radioactivity (Ci)*	6	--	--	4,600	--	54	***

*Non-entries, indicated by dashes, represent amounts that are identifiable but are insignificantly small.
**Applies only to uranium-recycle option.
***Decommissioning values not determined.



4.7.1 Treatment and Packaging

4.7.1.1 Technology Selection

Technologies for onsite management of low-level wastes exist and are routinely used. They are described below.

4.7.1.2 Description

Low level wastes (LLW) generated in the LWR fuel cycle fall into three categories: wet solid wastes, and compactable and noncompactable dry wastes. Wet solid wastes arise during the treatment of process streams to maintain plant water quality and to reduce radioactive concentrations in liquid effluents to acceptable levels. The principal liquid treatment processes are filtration, evaporation, and demineralization. These operations give rise to wet solid wastes such as filter sludges, evaporator bottoms, spent resins, and concentrated demineralizer regenerant solutions. Compactable dry wastes include such materials as contaminated rags, clothing, paper, ventilation air filters, and charcoal adsorbers, which are commonly pressed into 55-gallon drums for shipment to a licensed burial ground. Small noncompactable items such as discarded tools and laboratory glassware are usually included with these wastes.

The highest radioactive concentrations ($\geq$ 1 Ci/ft^3) in wet solid wastes are associated with filter/demineralizer sludges and spent demineralizer resins from a BWR water cleanup system, and spent filter cartridges and demineralizer resins from the letdown system of a PWR. The technology selected for handling these wastes is encapsulation in either a concrete or urea-formaldehyde matrix in a ratio of 2 parts waste to 1 part binder, and packaging in a DOT-approved 55-gallon drum for offsite

shipment. External shielding is supplied on transport vehicles if required to meet DOT dose-rate limits. This container was chosen for reference because it is considered to have a larger environmental impact in a transportation accident or in the event of container failure before burial than large (50-200 ft^3) containers with integral shields.

Filter sludges and spent bead resins from reactors and other fuel-cycle facilities that have a lower specific activity than reactor cleanup resins are commonly dewatered and shipped offsite in 55-gallon drums without addition of a solidification agent. Systems are available to solidify these wastes, but shipment of dewatered resins and sludges is selected as the reference technology since it has the potential for greater environmental impacts and is current industry practice.

Evaporator bottoms consist principally of concentrated (7% to 14%) boric-acid solutions from a PWR and concentrated (approximately 25%) sodium-sulfate solutions from BWRs that regenerate demineralizer resins. The latter are usually encapsulated in a cement or urea-formaldehyde matrix and shipped offsite in 55-gallon drums. Because of the difficulty of solidifying cement and urea-formaldehyde when mixed with concentrated boric acid solutions, it is common practice to ship boric-acid solutions in 55-gallon drums in an absorbent such as vermiculite. This is selected as the reference packaging technology.

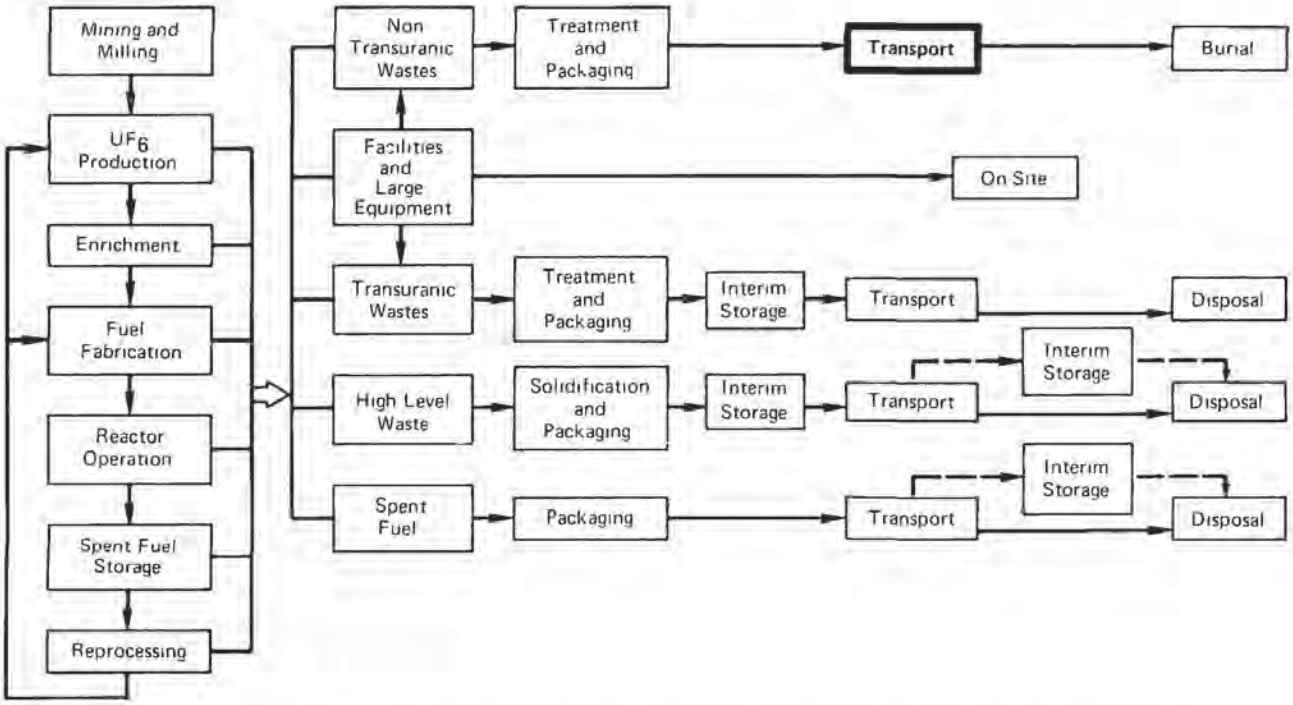
Volume-reduction systems are now available to reduce the quantities of wet solid wastes shipped from LWRs. A report describing one of these systems, which uses fluid-bed calcination, has been reviewed and found acceptable by the NRC staff.[2] Up to this writing, no licensee has proposed to include such a solid-waste volume-reduction system in his plant design. Therefore, this technology was not considered in the staff's evaluation.

Discarded equipment from fuel-cycle facilities with low fixed radioactivity is usually decontaminated to remove surface activity and thus facilitate handling and packaging. These items are then packaged in wooden crates for shipment offsite. Items from reactors with high induced radioactivity, such as control-rod blades, fuel-channel pieces, in-core instrumentation, and the like are sufficiently radioactive that they must be stored underwater in the spent-fuel pool to allow short-lived radioactive materials to decay before shipment offsite. These materials are cut up, if necessary, in the pool and shipped in a shielded container to the burial facility.

4.7.1.3 Environmental Impacts

The waste treatment and packaging systems to be installed at the various facilities are part of the process systems discussed in the applicant's Safety Analysis Report and Environmental Report. These documents contain analyses of the treatment and packaging systems and an estimate of annual radioactive release from the facility. The NRC staff's Environmental Statement for these facilities contains an analysis of the impact that results from operation of the facilities, including the treatment and packaging operations. Therefore, the environmental impacts of treatment and packaging are not addressed

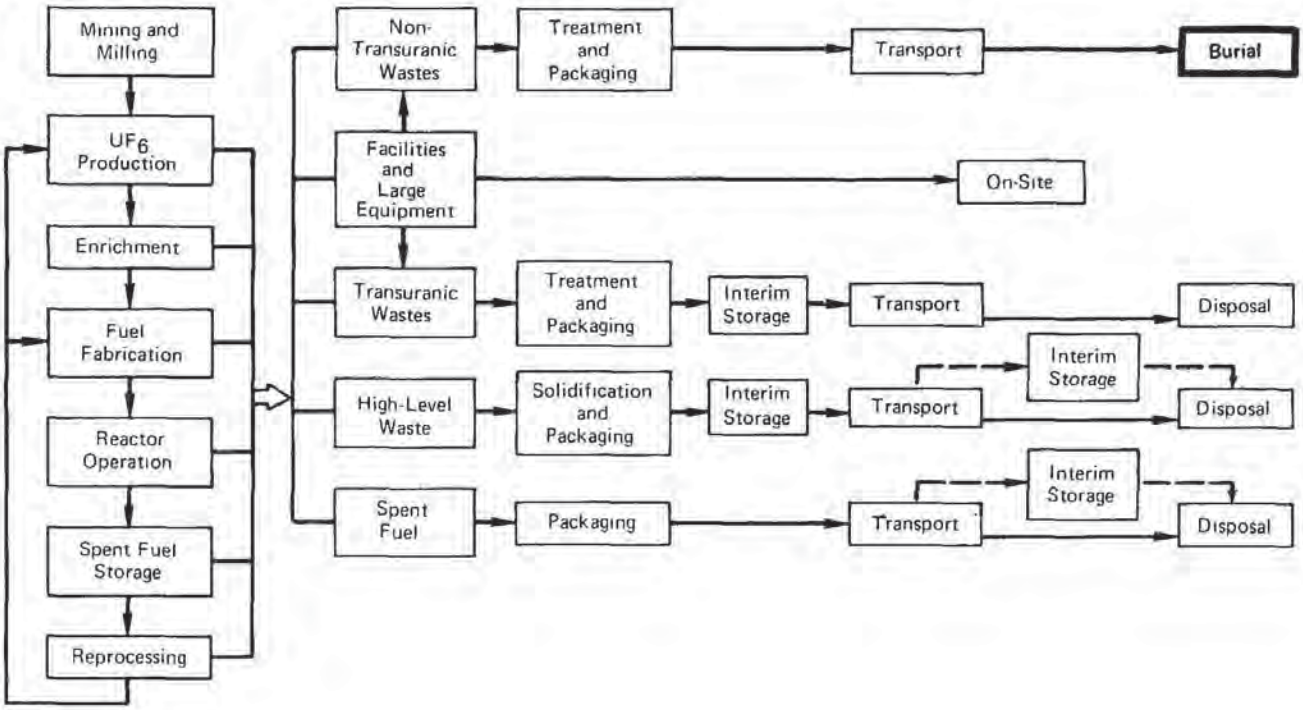
here. Solid wastes that are byproducts of facility operation are packaged and shipped to a licensed burial site in accordance with DOT and NRC regulations.[3,4]



4.7.2 Transportation

Transportation of LLW from all sources to the burial ground is described in Section 4.9. The environmental impact of these shipments, including normal and accident situations, is negligible from both radiological and nonradiological viewpoints.

3



4.7.3 Disposal

4.7.3.1 Technology Selection

The technology selected for discussion here for LLW disposal is shallow land burial. It is relatively inexpensive compared with other options, and there has been considerable experience during the past 20 years at both ERDA and commercial facilities.[5] This experience provides a basis for a technology description, and for analyzing the environmental impact.

There are six commercial sites licensed by NRC or its Agreement States for the handling and disposal of LLW,[5] and this number is expected to increase as existing sites reach full capacity. Environmental impacts of such future sites will be analyzed on a site-specific basis. Present sites receive a portion of their waste from the nuclear fuel cycle, and the rest from hospitals, universities, ERDA, and other sources. The fraction of waste originating within the nuclear fuel cycle is expected to increase in the future.[6]

Other options exist for the disposal of wastes in this category. Geologic disposal, discussed in Section 4.4, could be used. However, the primary advantage of geologic disposal is that it can isolate wastes from man's environment for hundreds of thousands of years, and such isolation is not clearly required for LLW.

A second disposal option is to place wastes in the sea, generally by dumping waste containers in the deep ocean basins. This practice is not now used in the United States, but is used by some nations that do not have sufficient appropriate land areas for shallow land burial.

4.7.3.2 Description

Simply described, model shallow land burial consists of excavating trenches 20-30 feet deep, filling them with wastes to within about 5 feet of the surface, and back-filling them with material removed during excavation.[7] This overburden provides radiation shielding for wastes with high gamma activity, prevents significant contact of plants and animals with the waste, is compacted to minimize water infiltration, and provides a buffer against waste exposure by erosion. Location and operation of the site are designed to restrict the amount of water moving through the waste material, to ensure that radionuclides will not migrate a significant distance from the burial trench.

Selection Criteria

The successful isolation of waste materials by shallow land burial requires careful site selection and proper management during both operational and post-operational phases.[8] The site must be generally devoid of surface water, located in a topographically high area where flooding is not possible, and be in a region of low seismic activity. The land surface should be generally flat, and sufficiently stable that rapid erosion will not occur. The ability of the soil or rock to transmit water should be high relative to the amount of water percolating into the ground from rainfall or snowmelt. This will generally prevent the buried waste from becoming saturated, thereby significantly reducing the quantity of radionuclides leached from the waste. The shallowest regional water table should be well below the bottom of the burial trench. The travel time of radionuclides that might be leached from the waste should be sufficiently long that radioactive decay will occur before the nuclides can enter offsite groundwater or surface-water systems in significant quantities.

The containment capability of the site can be further assured by proper management.[5] Surface drainage-control structures and grading will reduce water contact with the waste when trenches are open. Mounding of the final cover above the original land surface, compacting of the cap, and reseeding with sod-forming shallow-rooted vegetation, will minimize erosion and plant-root contact with the waste.

Finally, a substantial environmental surveillance program must be conducted. This program would require regular sampling of the air, surface water, groundwater, soil, plants, and animals within and around the site for measurement of concentration of radionuclides or other potentially toxic materials. If comparison of these concentrations with earlier ones (including levels observed before site operation) reveal significant changes, further investigation and possible corrective action will be undertaken.

Upon termination of burial activities at the site, final surface grading over the trenches, with revegetation as needed, will ensure good drainage with minimal erosion. Trenches will be marked with durable monuments of stone or metal bearing information about the volume of waste buried there and its radioactive content.

4.7.3.3 Bases for Impacts

Commercial and major ERDA burial grounds are located in nine states with a wide range of environments.[5] Operations at these sites are similar, varying somewhat with local environments and waste types. The model site described below encompasses characteristics of all the sites but is more representative of those in the humid eastern states.

The model land-burial facility for LLW is located in a rural, sparsely settled area. A central portion of this area, about 100 acres in size, is designated for waste handling and burial, and is fenced to prevent human and wildlife access. This restricted area is purchased by the site operator and deeded to the state. A surrounding buffer zone provides physical isolation of the central restricted area. The site is near a highway, and is located in an upland area, with no permanent surface water onsite. Surface drainage is good, and no portion of the site or its immediate surroundings has a potential for abnormal erosion. The regional water table is about 50 feet below the land surface, and fluctuates only a few feet annually. Mean annual precipitation is about 40 inches. Trenches are excavated to about 30 feet in unconsolidated materials of a silty texture. Permeability of these materials is low, but sufficient to allow vertical subsurface drainage of precipitation that percolates below plant-root zones. Thus, there are no major zones of perched water between the surface and the shallowest regional water table. The unconsolidated material is sufficiently uniform that measurements of water and radionuclide contents at selected monitoring points can be used to infer conditions between those points. The material is of sufficient adsorptive capacity that most radionuclides which might be leached from the waste by percolating water will move slower than the water.[9] There is no nearby use of groundwater downstream from the site. Seismic activity in the region is sufficiently low as to preclude significant earth movements.[10]

Buildings on the site provide space for offices, a laboratory, temporary storage for packaged radioactive wastes, and other necessary waste handling such as solidification of low-concentration liquids. Space is also available for storage of vehicles, earth-moving equipment, forklifts, etc., needed for trench preparation and handling of waste packages.

The site operator conducts an environmental surveillance program in accordance with state or federal requirements. The program includes sampling of air, water, soil, vegetation, and animals, both on and offsite, to determine whether radioactive material has migrated from the burial location. Surveillance records are maintained and submitted to the supervising agency. When burial operations terminate at the site, a surveillance program will be continued by the state or other responsible agency.

The facility keeps temporary storage of waste packages to a minimum. Packages are placed in a trench and covered with earth as soon after receipt as possible to minimize access to the waste, provide radiation shielding and fire protection, and minimize water entry. The trenches have water collection facilities for sampling and removal of accumulated precipitation. Such water, if contaminated above permissible release limits, is solidified onsite and buried. Completed trenches are revegetated to minimize soil erosion, and are marked as to location and contents. Areal dimensions

and spacing fo trenches is such that approximately 10,000 m^3 of waste can be disposed of per acre.[5]

Normal operating procedures at the site restrict the spread of contamination from the waste. However, the site operator employs trained health physicists to monitor all vehicles and personnel leaving the burial area to ensure that they are contamination-free. This procedure guards against the spread of contamination resulting from an abnormal incident. People entering the area are required to wear dosimeters which are checked to determine the magnitude of any possible dose received in the restricted area.[11]

Analytical Overview of Possible Impacts

Environmental impacts, both radiological and nonradiological, arising from the use of shallow land-burial sites can be divided into two categories: those associated with operational activities, and those resulting from natural phenomena during and after the operational period. Possible environmental effects resulting from site operations include land disturbance, alteration of surface drainage patterns, interference with plant and animal habitats, and changes in environmental quality resulting from the receiving, handling, and burying of waste. Possible impacts from natural phenomena include land erosion and the movement of nonradioactive materials from trenches into the environment by physical or biological processes.

The careful monitoring of personnel and vehicular traffic in and out of the area where radioactive materials are handled effectively eliminates the possibility of contamination being inadvertently carried offsite by such traffic. Consequences of advertent removal of contaminated materials are discussed in Section 4.10. Possible mechanisms by which the offsite environment could be affected include radionuclide transport in the air, water or animals crossing site boundaries, or direct radiation from waste materials. Any transport process must be preceded by an event or process which releases waste materials from containers or trenches.

4.7.3.4 Environmental Impacts

Resource Use

Land: - Site operations will only involve a portion of the total acreage at any one time. The restricted area, containing waste-handling facilities and disposal trenches, will be expanded as operations proceed, eventually encompassing the full 100 acres planned for trenches. During this operating period, access to the surrounding buffer zone can be controlled by stock fences and appropriate signs. Thus the impacts, visual and other, on the offsite land area will be minimal.

Before termination of activities at the site, all radioactive materials will be removed from the surface, trenches backfilled and graded, and the land revegetated. At that time there will be no potential for direct radiation exposure within or adjacent to the restricted area, so that the buffer zone can be released for other purposes. The land containing the burial trenches is assumed dedicated to waste disposal throughout the hazardous lifetime of the contained wastes. The total annual waste volume of 1100 m^3 generated by the model LWR fuel cycle (Table 4.29) will consume approximately 0.1 acre of land in this fashion per year.

The site will be surveyed before its use for the presence of any endangered plant or animal species and appropriate action taken to protect them. During the operational and post-operational periods, access to the restricted area and buffer zone will be controlled.

Water: - The consumption of water will be minimal, similar to that required by any small office complex. There will be no liquid surface effluents other than precipitation runoff. Drainage will be controlled to minimize sediment loads associated with runoff. Sewage will be treated using a septic tank or similar facility. The movement of radionuclides that might be leached from buried waste will be sufficiently slow as to permit their decay to innocuous levels before they can enter surface water bodies or the nearest groundwater supply.[4] Should surface contamination incidents occur during site operation, cleanup activities will ensure that no significant amount of radioactive materials will be transported offsite by surface runoff or atmospheric resuspension.

Energy: - The operation of earth-moving equipment, site vehicles, and office buildings will consume a small quantity of petroleum and electrical energy. This consumption is small compared with other waste-handling steps such as transportation.

Radiological (Operational Period)

Routine monitoring by the site operator is designed to prevent contamination from being inadvertently removed from the site by personnel or vehicular traffic during normal operations. Further, the burial site is designed to operate in such a fashion that under normal conditions, no detectable quantity of radioactive materials can move from burial trenches by air or surface water.[12] The waste is stored above ground for only brief periods, allowing minimal opportunity for either direct radiation exposure to the offsite environment, or for contact by small animals.

Radionuclide releases to the environment could result from a variety of accidental or abnormal situations. Data from commercial burial sites concerning such accidental operational releases are not extensive, but experience from ERDA sites can be used to estimate their significance. At Savannah River Laboratory (SRL), abnormal operating conditions are defined as "operational incidents that are a result of equipment malfunction or operator error. These events could cause a release of several curies of radioactivity, but no offsite effects would be expected."[13] During 20 years of operations at SRL, during which a waste volume equivalent to that produced by 200 RRY was buried, 68 incidents have occurred that could also occur at commercial sites.[13] These involved the rupture of packages before burial, spills of liquid waste, surface flooding, vegetation uptake, intentional exhumation of buried waste, and fires. Over 90% of these resulted in releases which did not extend beyond the burial-site fence. The remaining 10% resulted in contamination of small areas beyond the fence.[13]

Applications of the ERDA experience to commercial sites must take into account differences in operations at the respective sites. For example, waste packaging at

SRL need only be sufficient for onsite transport. Waste arriving at commercial sites will be packaged according to DOT specifications.[3] Wastes packaged to these more stringent requirements have a reduced potential for spillage or release during both normal handling and accident situations. Accidental opening of a waste package before burial and release of package contents could occur at a commercial burial facility. Normally, scattered waste materials would be confined to the immediate vicinity of the accident within the fenced facility and environmental impact would be negligible, especially if the wastes were solidified by incorporation in cement, urea-formaldehyde, or bitumen. The waste materials can be recovered and repackaged in drums. If it is assumed that the waste is readily dispersable and as much as one-third of the waste contained in a 55-gallon drum is released to the nearby unrestricted area, environmental effects would be insignificant. The volume of a 55-gal drum is approximately 7.4 ft^3. Assuming that the specific activity of the waste is 0.2 Ci/ft^3, the total amount of dispersed radioactivity would be about 500 mCi.

If the waste were uniformly dispersed over 1 acre, radioactive concentration would be about 11 $\mu Ci/ft^2$. This concentration is low; about 3 times higher than that in a 1-cm-thick layer of uranium ore.* Dispersion over a greater area would result in a proportionally lower concentration. The dispersed material would be in the vicinity of the site and could be recovered and repackaged.

Fires at SRL have occurred in drums of pyrophoric material, and in exposed waste in open trenches. These fires have not resulted in any offsite contamination. The likelihood of a fire at a commercial site is significantly less than for ERDA sites because of the DOT packaging requirements and site restrictions on explosive or pyrophoric materials. Atmospheric transport of radioactivity due to fires is primarily limited to the burning period. Analysis of the maximum consequences of the burning of an entire drum of waste indicates that potential doses to an individual at the site boundary are less than 1% of established guidelines.[13]

At the model site, abnormally high precipitation could result in water accumulation in open trenches. However, subsurface drainage at the site is sufficient that trench overflow will not occur. Water collection facilities in the trench can be used for sampling and/or removal of this water if it is sufficient to interfere with burial operations. Analysis will determine whether the water can be safely discharged or if it must be solidified for burial.

It is concluded that the environmental impacts of normal and accidental occurrences during burial operations are small.

Radiological (Natural Phenomena)

Burial-site operating procedures are designed to restrict the offsite transport of radionuclides by air or surface water during the operational period. Once operations terminate and all contaminated material has been buried, some release process

*The hazard index of uranium ore is about 10 times higher than that of the accumulated non-TRU wastes on a gross curie basis because of the long-lived alpha nuclides in the ore.

is necessary before transport in these two media can occur, such as penetration of the waste by plant roots or animals, erosion of surface cover material, or intentional or inadvertent excavation of the waste by man.

The principal radionuclides in LWR waste that will be buried have half-lives of 30 years or less. Applying a nominal period of 10 half-lives to allow decay to 0.1% of original activity, a period of 300 years is required to eliminate significant hazards to man or the environment. Thus, the containment capability of the model burial site must be evaluated for hundreds of years.

The minimum depth of uncontaminated cover material on burial trenches is 5 feet, with greater thickness in some places. Plant roots and activities of burrowing animals are commonly restricted to this depth, minimizing the possibility of contamination release by this process. The land surface in most areas undergoes continual erosion by wind and water. The erosion rate at the model site is estimated to be less than 6 inches per thousand years, typical of the eastern United States.[14] Site locations, surface grading, and revegetation will ensure that no abnormal erosion occurs. Thus, there will be no significant change in the depth of the cover material during the required isolation period.

Markers showing the location and contents of the burial trench will provide a warning against inadvertent excavation of the area. Probabilities of someone ignoring such markers cannot be estimated.

The leaching of radionuclides by percolating soil moisture and their consequent migration in the subsurface may occur during both the operational and post-operational periods. The rate at which these processes occur is directly related to the quantity of water contacting the waste. Therefore, the site operator conducts a drainage maintenance program designed to restrict surface runoff into open trenches. As mentioned, water collection facilities are present to move any significant volumes of water entering the trench if this should become necessary. The final cover material is compacted, using heavy equipment, to minimize water entry into the completed trench. The condition of the final cover is inspected regularly to identify any settling that might result from compaction of waste or soil. Such compaction is greatest within the first few years after trench completion.

These practices, described for the model site and generally followed at current sites, have resulted, in part, from experiences at existing burial sites. Burial sites located in relatively impermeable material, in regions of high precipitation, require particular attention to water control measures.[7] Water accumulation in trenches was a direct contributor to movement of radionuclides at existing sites at West Valley, NY, and Maxey Flats, KY.[15,16] (The West Valley site is not presently receiving waste for burial). Extensive water collection and treatment systems were required at both sites to handle accumulated trench water.[15,16] Radionuclides at concentrations above background have been detected in samples collected both on and offsite at the Maxey Flats facility. However, analysis by the NRC indicates that these elevated levels do not present any health or safety hazard to the public.[17]

These previous experiences must be treated as abnormal occurrences in the context of present burial technology. The environmental conditions at the sites differ substantially from those at the model site. Experience with water accumulation has led to modified water control measures at the sites, as expressed in the operating and maintenance procedures described herein. Environmental monitoring data collected within and adjacent to an operating burial site similar to the model site indicate that radionuclide releases to the environment can be prevented.[12]

Waste going to burial grounds is derived from many non-fuel-cycle sources.[6] Historically, these wastes contained long-lived TRU such as plutonium, which has caused concern about the long-term containment capability of burial sites. Present policies prohibit the burial of plutonium,* but some remains in place from previous years.

A reassessment of waste management regulations by the NRC is in progress, including the development of criteria for site selection. Preliminary conclusions indicate that combinations of improved site engineering, waste management, and packaging and solidification of wastes can minimize migration from the site.

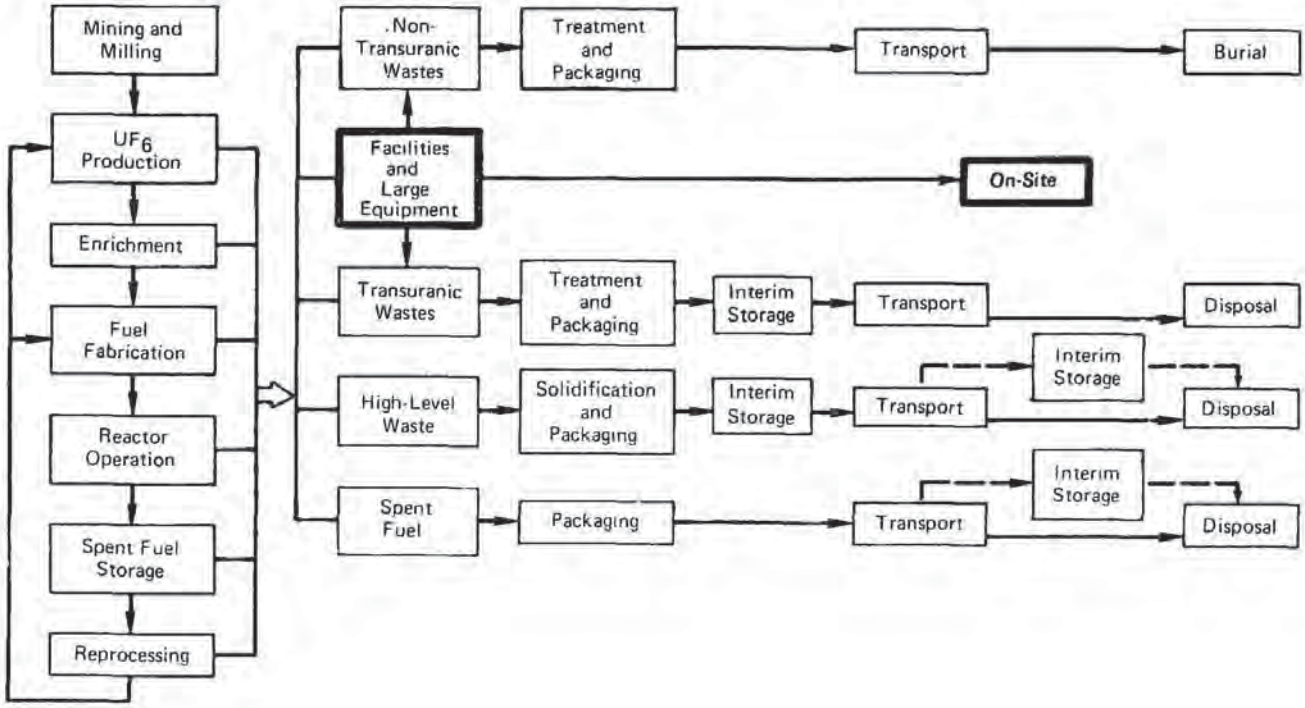
The benefits and disadvantages of potential modifications of current practices are presented in an ERDA survey.[5] The NRC staff will evaluate this survey and factor it into development of appropriate regulations and standards.

If further investigations indicate that corrective actions are necessary, several alternative actions are possible,[19] ranging from minor actions like surface grading and drainage modifications to complete exhumation of the waste. Experience indicates that the latter, while possible, poses many problems of potential environmental contamination.[20]

Air Impacts

There are no routine atmospheric effluents from the model site. Exhausts from vehicles and earth-moving equipment will not contribute significantly to altering air quality within or adjacent to the site. Pressurized gas containers are not accepted for burial, so there is minimal opportunity for gaseous releases from the waste before or after burial.[12] Organic decay processes within the waste material after burial can be anticipated, as well as generation of some gaseous radiolysis products. However, the low production rate is not expected to significantly affect air quality in the site environs. Earth-moving activities associated with trench construction and backfill, as well as vehicle travel in the site, will result in some atmospheric dust. However, control of surface contamination effectively eliminates the spread of radioactivity from this source.

*Except at the Hanford site.



4.8 Decontamination and Decommissioning of Nuclear Facilities

4.8.1 Introduction

This section presents estimates of environmental impacts from the decommissioning of light-water reactors, UF_6 production, enrichment, fuel fabrication and reprocessing plants, and spent-fuel storage facilities. There has been little experience in decommissioning commercial fuel-cycle facilities, so data on the magnitude of these impacts are not available. Decommissioning is not a technical problem, however, since the techniques have been demonstrated at reactors and government-owned facilities (see Annex 4.8.A). Most of the impact magnitudes presented in this section are staff estimates based on this experience. Preliminary results from NRC studies underway at Battelle Pacific Northwest Laboratories on the decommissioning of reprocessing plants and reactors, as well as discussions with industry personnel knowledgable in operation of various facilities, were used in making these estimates.

Most decommissioning experience to date has come from the decommissioning of nuclear reactors and critical facilities. Since 1960, 5 licensed nuclear power plants, 4 demonstration power plants, 6 licensed test reactors, 28 licensed research reactors, and 22 licensed critical facilities have been decommissioned.[1] All the decommissioning modes discussed below have been demonstrated at one or more of these facilities.

4.8.2 Decommissioning Alternatives

A variety of options is available for decommissioning nuclear facilities. The method chosen for a particular plant depends on several factors which are in many cases site-specific, depending on the planned use of the site after decommissioning. Future site use generally falls into two categories--restricted use and unrestricted use.

Under restricted use,* some portion of the site is usually reserved for "nuclear use" only. The public is denied access to contaminated portions of the facility. Routine surveillance is conducted to verify that radioactive material does not escape through containment barriers. A license is maintained by the facility owner.

For unrestricted use,* the facility license is terminated and the site may be abandoned or released for unrestricted use. Uncontaminated portions of the facility may be left on the site.

Alternative decommissioning modes are:

- Layaway
- Protective storage
- Entombment
- Decontamination

In the layaway mode, the public and the environment are protected by the methods employed when the facility was in operation, such as building structural integrity and filtered ventilation systems. Appropriate security measures are taken to limit access to the site and facility. All safety systems are monitored continuously while the facility is in layaway, and a surveillance and maintenance program is carried out to ensure that the facility continues to provide adequate public safety. If long-lived radioactivity is present in the facility, layaway is not expected to be a permanent decommissioning mode. Further decommissioning will be required.

To place a facility in protective storage, temporary physical barriers are erected between the environment and portions of the facility that have radiation or contamination levels exceeding those specified in Regulatory Guide 1.86.[2] Appropriate security measures are used to restrict public access to the site. A surveillance and maintenance program is established to monitor the integrity of the physical barriers and security systems.

In the entombment mode, the public is protected from radioactivity remaining at the site by the erection of permanent physical barriers. These barriers are engineered to remain in place until radioactivity has decayed to levels that permit unrestricted use of the site. A surveillance program is established to monitor the integrity of the barriers.

In the decontamination mode, all potentially hazardous amounts of radioactive material are removed from the site. Decontamination of the facility permits the owner to apply to NRC for termination of the facility license.

Layaway has generally been used at sites where other nuclear facilities remain in operation after the reactor is retired. The availability of full-time security, maintenance, and surveillance personnel at government-owned sites, coupled with relatively low costs, make layaway an attractive option under these circumstances.

*See 10 CFR 20 and 20.3 for definitions.

Layaway has been used to decommission the Hanford production reactors,[3] the Fermi Reactor,[4] Peach Bottom,[5] Vallecitos,[6] GEEVSER,[7] Westinghouse Test Reactor,[8] and the NASA Plumbrook Reactor.[9] The Redox reprocessing plant at Hanford has also been placed in layaway.[10]

The principles involved in placing a reactor in protective storage have been demonstrated at several facilities including CVTR,[11] EBR I,[12] SEFOR,[13] Saxton,[14] and Pathfinder.[15] The costs of placing a facility in protective storage are generally somewhat higher than for layaway, but surveillance and maintenance costs should be lower. The costs of placing a reactor in layaway or protective storage have been estimated at about 10-20% of the costs of completely decontaminating a facility.[16]

The entombment mode has been used to decommission the Hallam,[17] Piqua,[18] and Bonus[19] reactors. Entombing costs are significantly higher than for layaway or protective storage, but surveillance costs will be minimal. Entombment costs for a reactor have been estimated to be about 50% of decontaminating costs.[16]

Decontamination has been demonstrated at the Elk River Reactor[20] and at several small test reactors and critical facilities.[1] There has also been experience in the decontamination of government-owned plutonium-contaminated facilities.[21-25] The capital outlay for decontamination is higher than for the other disposition modes, but there are no continuing costs or responsibilities when decontamination is complete. The cost of decontaminating a reactor or reprocessing plant has been estimated at 10-20% of original facility costs.[16]

4.8.3 Selection of a Decommissioning Alternative

The decontamination mode has been selected for estimating the environmental impact attendant upon decommissioning of nuclear facilities. The reasons for the selection are:

- Decontamination is the most conservative of the options since it is the most extensive treatment of a facility and as a consequence could be expected to have the greatest potential environmental impact.

- There is little incentive to choose one of the other disposition modes for facilities at the front end of the fuel cycle. Radiation exposure levels are generally so low that equipment can be removed by normal maintenance procedures.

- Decontamination places radioactive wastes in fewer centralized locations. The number of sites at which surveillance activities must be performed is thus greatly reduced over decommissioning alternatives that leave radioactivity at individual plant sites.

- Monetary costs of decommissioning are borne immediately by the facility owner when the facility is decontaminated.

- Disposition modes that rely on long-term surveillance and maintenance at the site require corporate survival of the facility owner or assumption of these activities by the government.

- If entombment were selected for facilities that contain significant amounts of long-lived radionuclides such as plutonium, barriers between radioactivity and the environment would have to be maintained for very long periods of time.

- Decontamination releases the site for unrestricted use. Other disposition modes restrict at least portions of the site for long periods of time.

- Decontamination to unrestricted-use levels permits the plant owner to apply to NRC for termination of the facility license.

4.8.4 Summary of Environmental Effects of Decommissioning

The estimated environmental impacts of decommissioning nuclear facilities are summarized below. Table 4.30 presents the estimated total impact of decommissioning a particular plant and Table 4.31 shows these impacts normalized to one year's operation of a 1000-MWe reactor for throwaway and uranium-recycle-only, respectively.

The most significant potential environmental effects associated with decontamination of a nuclear facility include:

- Costs of Performing the Operations. These costs are expected to range from 1 - 2%[26] of original plant construction costs for some facilities, up to 15 - 20%[16,27] for other facilities.

- Contaminated Waste Disposal. A variety of wastes are produced when a plant is decontaminated. They include contaminated process equipment and piping, concrete rubble and other structural materials, stainless-steel liners from process cells and storage pools, and generally combustible trash such as rags, gloves, protective clothing, and plastic bags used in decontamination operations. Wastes from decontaminating a reactor include a large volume of materials and equipment contaminated to relatively low levels, and a smaller volume of components and structures with higher levels of activation. The latter will be segmented and transported in shielded shipping containers. The wastes from the head-end, plutonium dioxide conversion, and HLW solidification portions of the reprocessing plant are assumed to be TRU wastes that will be disposed of at a federal respository. All other contaminated waste will be disposed of at commercial burial sites. Contamination levels of wastes transported from decontaminated facilities are expected to be generally low. Most could be transported as low-specific-activity material and would not require radiation shielding to meet the dose limitations of 49 CFR 173. Possible exceptions could be equipment and contaminated structures removed from certain high-activity operations, such as the head-end cell and

Table 4.30

ENVIRONMENTAL IMPACTS FROM DECOMMISSIONING OF NUCLEAR FACILITIES (PER FACILITY, NORMAL OPERATION)

Natural Resource Use	UF_6 Production	Enrichment[a]	UO_2 Fuel Fabrication	Reprocessing	Spent Fuel Storage	LWR Reactor[d]
Water (Millions of gallons)						
Discharged to air	7.5		1.2×10^{-1}	1.1×10^{-1}	-	1.1
Discharged to water bodies	-		6.0×10^{-2}	3.0	8.2	-
Sanitary Water	3.2×10^{-1}		5.6	4.7	4.6	4.9
TOTAL WATER	7.8		5.8	7.8	12.8	6.0
Electrical Energy Use	b		b	b	b	b
Radiological (Curies)						
Gases						
Ra 226	9.9×10^{-4}		-	-	-	-
Th 230	9.9×10^{-4}		-	-	-	-
Uranium	7.4×10^{-3}		5.1×10^{-3}	-	-	-
Fission Products	-		-	5.0	-	-
Transuranics	-		-	5.0		-
Liquids						
Uranium & Daughters	-		7.0×10^{-3}	-	-	-
Gross beta & gamma	-		-	3.4×10^{-4}	9.3×10^{-4}	1.3×10^{-4}
Solids (ft^3)[c]						
Non-TRU						
Combustible	1.5×10^{3}		2.5×10^{3}	3.8×10^{4}	5.0×10^{3}	2.0×10^{4}
Non-Combustible	5.1×10^{4}		2.4×10^{5}	2.8×10^{5}	1.0×10^{5}	3.5×10^{5}
TRU						
Combustible	-		-	3.3×10^{4}	-	-
Non-Combustible	-		-	1.4×10^{5}	-	-
Occupational Exposure (person-rem)	<83		<63	400	50	1.1×10^{3}
Population Exposure	b		b	b	b	

a. The impact of decommissioning enrichment plants has not been evaluated (see text and Annex 4.A).

b. Decommissioning values are estimated to be less than or equal to the impact during normal facility operation.

c. Packaged in 200-liter drums or fiberglass-reinforced plywood boxes. A packing factor of 60% is assumed.

d. Based on PWR since impact is slightly higher than for BWR.

Table 4.31

SUMMARY OF ENVIRONMENTAL IMPACTS FROM DECOMMISSIONING NUCLEAR FACILITIES (NORMAL OPERATIONS)
(Normalized to Model LWR Annual Fuel Requirements)

Natural Resource Use	U-Only Recycle	No Recycle ("Throwaway")
Water (Millions of gallons)		
Discharged to air	3.4×10^{-3}	4.0×10^{-3}
Discharged to water bodies	3.9×10^{-2}	4.9×10^{-2}
Sanitary water	1.6×10^{-1}	1.6×10^{-1}
TOTAL WATER	2.0×10^{-1}	2.2×10^{-1}
Electrical Energy Use	(a)	(a)
Radiological (Curies)		
Gases		
Ra 226	4.5×10^{-7}	5.3×10^{-7}
Th 230	4.5×10^{-7}	5.3×10^{-7}
Uranium	7.3×10^{-6}	7.9×10^{-6}
Fission Products	2.9×10^{-3}	-
Transuranics	2.9×10^{-4}	-
Liquids		
Uranium & daughters	5.4×10^{-6}	5.4×10^{-6}
Gross beta & gamma	4.5×10^{-6}	5.9×10^{-6}
Solids (ft^3)		
Non-TRU		
Combustible	6.9×10^{2}	6.8×10^{2}
Noncombustible	1.2×10^{4}	1.2×10^{4}
TRU		
Combustible	19	-
Noncombustible	82	-
Occupational Exposure (person-rem)	37	37
Population Exposure	(a)	(a)

(a) Expected to be small on a normalized basis.

the HLW solidification hot cell at the reprocessing plant. Shielded shipping containers may be required for these materials. The impacts of transportation of decommissioning wastes is included in Section 4.7.2, and disposal of decommissioning wastes is included in Sections 4.3 and 4.7.

- Occupational Radiation Exposure. It is expected that most exposure during decontamination operations will result from periods of time spent in relatively low exposure areas (1-10 mrem/hr).[28] Operations in high-exposure areas (particularly in reprocessing plants) will be performed remotely using manipulators or long-handled tools.

- Public exposure. It is expected that public radiation exposure will be less during decontamination than during normal plant operation. Inventories of radioactive material in the plant during decontamination are much smaller than during normal operation, and most of what remains is material that is bound to equipment or structural surfaces. All safety systems (such as filtered ventilation equipment) remain in operation until most of the activity has been removed from the facility.

- Accidental Public Radiation Exposure. For many of the same reasons, public exposure to radiation from accidents during decontamination is estimated to be much lower than during normal operation. The consequences of an accident during decontamination are expected to be negligible since most radioactivity remaining at the facility is in a relatively nondispersable form.

- Nonradioactive Waste Disposal. Some nonradioactive materials will require disposal from some facilities. It is assumed that all process chemicals not required for decontamination will have been removed from the facility when decommissioning begins. Other uncontaminated materials can be sold for scrap or disposed of in a sanitary landfill.

- Resource Commitments. The major resource commitment in the decontamination of a nuclear facility is the commitment of land at a shallow burial site or space in a federal waste repository. This commitment is offset by release of the plant site for unrestricted use. Materials such as decontamination solutions, protective clothing, rags, special tools, and equipment are consumed during decontamination operations (i.e., they become contaminated and are disposed of as required by regulations).

- Water Usage. Sanitary water usage during decontamination is expected to be approximately the same as during plant operation. Water is also used in the preparation of chemical decontamination solutions. Most of this is evaporated to the atmosphere when the contaminated solutions are concentrated and the resulting wastes are solidified.

- Water Discharged. The decommissioning of some facilities may require the discharge of water bearing small amounts of radioactive contaminants to local water bodies. Radioactivity levels in discharged water will be required to meet the provisions of 10 CFR 20. Sources of such water are drainage from fuel and solidified HLW canister storage pools at reprocessing plants, from storage pools at independent spent-fuel storage installations, from washdown of contaminated equipment to relatively low radiation levels, from reactor storage pools, and distillates from chemical decontamination solutions.

- Other Impacts. Other impacts associated with decontamination are typical of a heavy construction project, including operation of heavy equipment such as cranes, bulldozers, and increased vehicular traffic. Electrical power consumed during decontamination is expected to be at a rate equal to or less than during plant operation.

A variety of sources was used to prepare the estimates of environmental impacts presented in Tables 4.30 and 4.31. In the case of reactors, actual decommissioning experience was used (see Annex 4.8.A). The information on fuel reprocessing plants is based on preliminary results from decommissioning studies underway at BNWL under NRC sponsorship. Less is known about decommissioning other fuel-cycle plants. The figures shown for these plants are NRC staff estimates, based when possible on discussions with industry personnel familiar with the actual plants.

Some simplifying assumptions have been made in obtaining the estimates in the tables. It is assumed that all feed and product materials and process chemicals that can be economically recovered will have been removed from the plant before decontamination begins. In cases where the information available was for a plant of a different size than the model, the results were scaled linearly to the model plant size. All facilities are assumed to be of the general type that will be built and operated in the future (for example, it is assumed that there are no onsite burial grounds). Annex 4.8.A gives a brief overview of the steps that would be used for decontaminating each nuclear facility. While it has not been possible to estimate the magnitudes of these impacts for enrichment facilities, it is believed that they will be low (see discussion in Annex).

ANNEX 4.8.A

DECONTAMINATION OF NUCLEAR FUEL CYCLE PLANTS

4.8.A.1 Fuel Reprocessing Plants

The steps outlined here for decontaminating an FRP are based on the preliminary results of an NRC-sponsored study underway at Battelle. The model plant used in the BNWL study is typical of reprocessing plants now being built or designed. For purposes of assessing the environmental impacts of decommissioning, the plant is alsc assumed to have facilities for converting plutonium nitrate to plutonium dioxide, for converting uranyl nitrate to UF_6, and for HLW solidification, with sufficient interim storage capacity for about five years' output from the solidification plant.

The facilities would be decontaminated in the following order (the first three could proceed concurrently):

- Process buildings (including PuO_2 conversion)
- UF_6 facility
- Fuel receiving and storage station
- Liquid HLW storage
- HLW solidification
- Solidified HLW storage
- Ventilation filter station

Decontamination of an FRP generally proceeds in three steps:

- Chemical Flushes. This step is designed to reduce radiation levels in areas where contact operations are required, and to reduce radiation exposure during packaging, handling, and shipping of radioactive wastes. Strong reagents such as $HF-HNO_3$ can be used, since equipment damage is not a consideration. After flushing, chemical decontamination solutions are concentrated using the FRP's process equipment, placed in HLW storage tanks and eventually solidified.

- Equipment Removal. In areas where radiation levels exceed 50-100 mrem/hr, equipment is removed remotely using installed equipment or specially designed equipment. Since most equipment and piping is stainless steel, portable plasma torches may be used for equipment removal. Large equipment is cut into smaller pieces to facilitate packaging and shipping. All equipment is shipped offsite to a commercial burial ground or a federal repository (TRU-contaminated waste).

- Removal of Stainless-Steel Liners and Contaminated Concrete. Plasma torches are used to cut stainless-steel liners in place into pieces convenient for shipping. Concrete is decontaminated by removing the outer three to four inches using explosives[1] or rock-splitters.[2] Contaminated concrete rubble is loaded into shipping containers using a front-end loader with a shielded cab.

Estimates of wastes from decontaminating the HLW solidification portion of an FRP are based on the conceptual design presented in Ref. 3. The facility consists of two hot cells about 30' long x 15' wide x 40' high and an operating gallery. One cell contains a waste concentrator, calciner, melter, furnace, canister welding station, canister test station, interim canister storage racks, and associated equipment. The other cell contains off-gas treatment facilities including evaporators, nitric acid fractionator, caustic scrubber, heat exchanger, filters, and associated equipment. Both cells have tools for remote operation and maintenance. The facility is also assumed to have a canister storage pool that can store approximately 5 years' output from the solidification plant. This pool is assumed to be similar in size and construction to the plant's spent-fuel storage pool.

Estimates of wastes from decontaminating the plutonium waste treatment facility are based on the Barnwell Nuclear Fuel Plant design of the plutonium product facility presented in Ref. 4. Although the processes may be different, the facilities in which they are carried out should be similar. The plutonium waste treatment facility has an annual design capacity of 25,000 kg of plutonium with a conversion capacity of 100 kg of plutonium per day using two conversion lines. The facility is located in an earthquake and tornado-resistant structure containing equipment, cells, and rooms for the following operations:

- Production of an impure plutonium oxide from plutonium nitrate feed
- Packaging of plutonium oxide powder
- Plutonium oxide storage
- Shipping container storage and refurbishing
- Interim storage of drummed waste.

Estimates of wastes generated from decontaminating the UF_6 facility at an FRP are based on the 1500 MTU/yr facility at the AGNS FRP, adjusted to the 2000 MTHM capacity basis used herein. The plant contains equipment for the following process steps:

- A concentration step that converts uranyl nitrate feed to uranyl nitrate hexahydrate (UNH)
- A concentration step that converts UNH to UO_3

- Reduction of UO_3 to UO_2 in a fluidized bed
- Production of UF_4 by reacting UO_2 with anhydrous HF in a fluidized bed
- Production of UF_6 by reaction of UF_4 with elemental fluorine in a fluidized bed
- Collection of UF_6 in cold traps.

The uranium feed material will be contaminated with trace amounts of plutonium and fission products. It is expected that some equipment will be contaminated with plutonium over the lifetime of the plant and will require disposal as TRU waste when the plant is decommissioned. The remainder of the equipment in the plant may be chemically flushed to acceptable radioactivity levels that would permit release for resale or scrap.

4.8.A.2 UO_2 Fuel Fabrication Plant

The environmental impact of dismantling a UO_2 fuel fabrication plant has been estimated with the assistance of knowledgable industry personnel.[5] Most of the equipment requiring removal in a decontamination operation is at the head end of the fuel fabrication process. It includes equipment for vaporization of UF_6 from the shipping containers, ammonium diuranate preparation, calcining, blending, scrap recovery (including scrap dissolution, liquid storage, solvent extraction and ammonium diuranate preparation), pelletization, sintering, and grinding. Some equipment will be chemically flushed and sold for use in other fuel fabrication facilities. The remainder would be packaged and shipped to a commercial burial ground. Chemical decontamination solutions will be fixed in a solid such as concrete and shipped to a disposal site. Some low-level aqueous wash solutions will meet the standards of 10 CFR 20 and may be discharged to local water bodies.

Some concrete surfaces in the facility will be contaminated to radioactivity levels exceeding those specified in Regulatory Guide 1.86. The contaminated surface of this concrete would be removed and the rubble packaged and shipped to a disposal site. The remaining uncontaminated structures are expected to be left in place.

All decontamination operations will be performed by direct contact methods.

4.8.A.3 Uranium Hexafluoride (UF_6) Plants

No commercially built UF_6 plants of the size currently operating or projected for operation have ever been decommissioned. One ERDA-operated uranium facility (Weldon Springs) was partially decontaminated by the army and its contractors in the 1960s.[6] Estimates of the environmental impact resulting from the decommissioning of a UF_6 facility must be primarily based on extrapolations from other decontamination and decommissioning activities involving similar source materials,[7] and on engineering analysis. The following assumptions were made to simplify the analysis:

- The model plant will employ the wet solvent extraction process. This process was selected since it is the most widely used (by 5 of the 6 UF_6 plants in the world) and because it has a greater potential for facility contamination.

- The model plant capacity is 10,000 metric tons of uranium compared to the 15,000-MTU GESMO.

- All solvent extraction raffinate wastes are removed from the plant before decommissioning begins.

- The plant is decontaminated to the levels listed in Table 4.32. These levels have been accepted by NRC for releasing similar facilities from licensing requirements under 10 CFR 40.

- The plant is shut down as a planned operation for decommissioning. An effort is made to minimize processing equipment inventories before shutdown.

- All plant areas except uranium processing areas are kept contamination-free during normal operation and so remain during decommissioning.

The general procedures for decommissioning the model UF_6 facility are as follows:

- Reusable or resalable materials or equipment that can be decontaminated below the recommended guidelines for the unlicensed use of such material will be decontaminated and salvaged. Unsalvable equipment will be disposed of through an authorized licensee.

- Equipment to be decontaminated will be water-washed or steam-cleaned. If further cleaning is necessary to meet decontamination guidelines, chemicals or abrasives will be used.

- Buildings are vacuum cleaned after process equipment has been removed, and washed down with high-pressure water. Chemical cleaning may be required in some areas.

- All liquids from the miscellaneous digestor and decontamination activity are collected and the purified product is sold.

- Liquid waste streams are treated in an evaporator and the solids are packaged for transfer to an authorized licensee.

Table 4.32

ASSUMED REQUIREMENTS FOR TERMINATION OF UF_6 PLANT LICENSE*

1. All chemical mixtures, compounds, solutions, or alloys containing 0.05 percent or more of source material by weight have been transferred to authorized licensees, and

2. Contamination on process equipment and buildings does not exceed the following:

   a. The maximum amount of fixed alpha radioactivity: 25,000 disintegrations per minute per 100 cm^2.

   b. The average amount of fixed alpha radioactivity: 5,000 disintegrations per minute per 100 cm^2.

   c. The maximum amount of removable alpha radioactivity: 1,000 disintegrations per 100 cm^2.

   d. The maximum radiation level 1 cm from the surface: 1 millirad per hour as measured with an open-window survey meter through a tissue equivalent absorber.

   e. The average radiation level 1 cm from the surface: 0.2 millirad per hour.

4.8.A.4 Uranium Enrichment Facilities

Existing ERDA enrichment facilities use the gaseous diffusion process. In the near future, gas centrifuge technology, with its reduced power requirements, may become sufficiently economical to provide the enriched uranium required for private industry.

The model gas centrifuge enrichment facility will produce 8.75×10^6 SWU per year. It will have numerous centrifuge machines operated in cascade and will be housed in eight buildings, each approximately 425 ft wide by 650 ft long. Associated process buildings will be equipped with feed stations plus product and tailings withdrawal facilities. There will be facilities for equipment assembly and maintenance including decontamination and scrap processing.

A gaseous diffusion plant consists of numerous diffusion stages operated in cascade, with each stage providing small incremental enrichment. Support facilities will be as above. A description of the diffusion and centrifuge processes and their associated facility designs can be found in Ref. 8.

*Based on Ref. 7.

Detailed information on uranium enrichment facilities is classified, and there is no published information on decommissioning enrichment plants. It has not been possible to make numerical estimates of environmental impacts associated with decontaminating either gaseous diffusion or centrifuge enrichment plants. Nonetheless, several factors lead to the preliminary conclusion that these impacts will be generally low:

- The primary contaminant in the facility is uranium, so radiation levels in the facility and on removed equipment will be low. Occupational exposure during decontamination should be low. Normal maintenance techniques are routinely used to remove the equipment.

- Both enrichment processes require high-integrity equipment. Contamination is confined inside the process equipment and piping.

- The UF_6 would be purged from the system at the end of plant life since it is a valuable product. Only residual contamination would remain. Much of the process equipment is constructed of high-integrity metals that can be readily decontaminated with chemicals such as nitric acid. Chemical decontamination could be expected to significantly reduce the amount of equipment requiring disposal at a burial site. Decontamination solutions would be concentrated, solidified, and transported offsite for burial.

4.8.A.5 Light Water Reactors

The environmental effects of decontaminating LWR's are based on staff estimates with the assistance of industry personnel experienced in reactor decommissioning. Reactor decontamination has been demonstrated at the Elk River Reactor[9] and at several small test reactors and critical facilities.[10] The B&W reactor was partially decontaminated[11] and the SRE reactor is currently being decontaminated.[12] The estimates presented here are based on current-generation PWR's and BWR's.

Decontaminating an LWR involves:

1. Draining and chemically flushing the primary system. Chemical flushes are designed to reduce occupational exposure during steam generator removal. Chemical solutions are concentrated and solidified. Wash solutions and primary water are demineralized and discharged to local water bodies.

2. Removing steam generators (PWR only) and primary pumps. The steam generators are segmented for shipment on a railcar or truck.

3. Removing reactor internals, which are segmented for shipment using an arc saw or plasma torch. Such operations can be performed remotely.

4. Removing reactor vessel and surrounding contaminated structures. The vessel would be removed with an arc saw (PWR) or a plasma torch (BWR). This operation is performed remotely. Activated concrete in the reactor cavity is removed with explosives.[1]

5. Removing remaining reactor building equipment. Contact methods are used.

6. Removing the turbine and decontaminating the turbine building (BWR only).

7. Removing contaminated equipment from auxiliary buildings.

4.9 Transportation of Radioactive Wastes

4.9.1 Packaging Technology

Other sections of this report describe the quantities of radioactive waste generated in the nuclear fuel cycle for the alternatives of no recycle and uranium-only recycle. Table 4.33 summarizes these quantities in terms of volume, and specifies the numbers of shipments per reactor they represent for each portion of the nuclear fuel cycle. Table 4.34 summarizes the characteristics, and specifies radioactivity per shipment, for each class of waste (low-level, transuranic, and high-level).

4.9.2 Transportation Description

The analysis in this chapter does not treat the transportation of fresh fuel to a reactor, or waste from it, except when spent fuel is considered in the no-recycle case. The impact of transportation to and from reactors is described in WASH-1238, "Environmental Survey of Transportation of Radioactive Materials to and from Nuclear Power Plants."[1] The analysis in this section treats the following instances of waste transportation associated with the nuclear fuel cycle:[2,3]

- Shipment of low-level non-TRU wastes from UF_6 plants, enrichment plants, uranium fuel fabrication plants, and reprocessing plants to commercial land burial sites;

- Shipment of TRU wastes from reprocessing plants to an interim storage facility, and then on to permanent disposal;

- Shipment of irradiated nuclear fuel from nuclear reactors (for the no-recycle case) to an interim storage facility, and then on to permanent disposal, and

- Shipment of HLW from spent-fuel reprocessing facilities to an interim storage facility, and then on to permanent disposal.

Most shipments of radioactive waste material related to the nuclear fuel cycle move in routine commerce and on conventional transportation equipment, although normally on an "exclusive-use" basis. Shipments are therefore subject to the same transportation environment, including accidents, as nonradiological cargo. Although a shipper may impose some conditions on the carriage of his exclusive-use shipment, such as speed limitations, providing an escort, etc., most of the conditions to which his shipment is subjected and the probability of its being involved in an accident are not, for the most part, subject to his control.

Protection of the public and transport workers from radiation during shipment of radioactive materials is achieved by a combination of (1) limitations on the radioactive contents of the package and shipments, and (2) standards and criteria for package design and control. An acceptable level of safety in transportation of fuel cycle materials is achieved without special routing, although such routings are sometimes used at bridges and tunnels to avoid interference with traffic should an accident occur.

Table 4.33

CHARACTERISTICS OF WASTE SHIPMENTS

Source	Type Waste	Trans. Mode.	Miles per Shipment	Volume/ Shipment (m^3)	U-Only Recycle (per reactor)			No Recycle (per reactor)		
					Waste Volume (m^3)	Shipments	Shipment Miles	Waste Volume (m^3)	Shipments	Shipment Miles
UF_6 Production	Low-Level	Truck	500	8.4	55.3	6.6	3290	53.5	6.4	3180
Enrichment Process	Low-Level	Truck	500	8.4	30.0	3.6	1790	28.0	3.3	1670
Fuel Fabrication	Low-Level	Truck	500	8.4	28.5	3.4	1700	28.5	3.4	1700
Fuel Reprocessing	Low-Level	Truck	500	8.4	7.02	.84	420	-	-	-
	TRU	Truck	1500[1]	8.0	29	3.6	5440	-	-	-
	High-Level	Rail	1500[1]	1.92	2.19	1.14	1710	-	-	-
Interim Storage Facility	TRU	Rail	500[2]	20	29	1.4	720	-	-	-
	High-Level	Rail	500[2]	1.92	2.19	1.14	570	-	-	-

[1] Transported to interim storage facility

[2] Transported to permanent storage

Table 4.34

SUMMARY OF WASTE SHIPMENTS

Type of Waste	FUEL CYCLE OPTION: U-Only Recycle — Approx. Shipments	Mode	Shipment Miles	Curies/ Shipment	Throwaway (No Recycle) — Approx. Shipments	Mode	Shipment Miles	Curies/ Shipment
Low-Level	14.4	Truck	7,200	23	13.1	Truck	6,550	23
TRU	3.6	Truck	5,440	14,000	-	-	-	-
	1.4	Rail	720	35,000		-	-	-
High-Level	2.3	Rail	2,280	1.5×10^6				
Totals - Truck	18.0		12,600		13.1		6,600	
Rail	3.7		3,000					

Primary reliance for safety in transport of radioactive material is placed on packaging. The packaging must meet regulatory standards established by the DOT, the NRC, and the states, according to the type and form of materials, for containment, shielding, nuclear criticality safety, and heat dissipation (see Ch. IV, Sec. G, App. A of GESMO[4] for a summary of important regulations and requirements). The standards for packages containing significant radioactivity provide that the packaging shall prevent loss or dispersal of the radioactive contents, retain shielding efficiency, assure nuclear criticality safety, and provide adequate heat dissipation under normal conditions of transport and under specified (hypothetical) accident damage test conditions.[5] The contents of packages not designed to withstand accidents are limited, thereby reducing the risk of accident releases. Package contents also must be limited so that standards for external radiation levels, temperature, pressure, and containment are met.

Protection from external radiation is provided by limitations on radiation levels at the surfaces of packages of radioactive materials, and by stowage and segregation provisions. The number of packages in a single vehicle or area is limited to control the aggregate radiation level and to provide nuclear criticality safety. Minimum separation distances from people are specified for loading and storing packages of radioactive material.

4.9.3 Bases For Impacts

Shipments/miles/mode - The number of shipments and shipment-miles are summarized in Table 4.34. Where truck is specified as the mode of transport, rail and barge are other conceivable modes, but the numbers of shipment miles would be less, resulting in a smaller overall impact on the environment. Where rail is specified, truck is not considered a conceivable mode, and the number of shipment miles may be somewhat greater than for rail. Because of the remoteness of barge shipment routes from the general public, however, the radiological impact of a barge shipment is considered to be always less than that of a rail shipment of the same waste.[6]

External Radiation Levels - Shipments of radioactive waste must meet DOT regulatory limitations on radiation levels outside the package and the transporting vehicle (49 CFR 173.393).[7] Shipments of nuclear waste are normally made in exclusive-use trucks and railcars. Special provisions of the DOT regulations therefore apply which limit radiation levels outside the transporting vehicle to 10 mrem per hour six feet from the edge of the vehicle.[7] For simplicity, it is assumed that all shipments produce radiation levels at this upper limit, even though this is unlikely, especially for low-level waste shipments. This amounts to radiation levels of about 16 mrem per hour three feet from the vehicle edge.[8] DOT regulations limit radiation levels in the truck cab to two mrem per hour. For this analysis it is assumed that the radiation level in the cab of a truck carrying HLW and TRU shipments is 2 mrem/hr. For low-level wastes, most drums will contain such small quantities of radioactivity that the radiation level in the truck cab will not exceed 0.2 mrem/hr.[9]

Radioactivity - The average radioactivity of fuel cycle waste shipments is summarized in Table 4.34.

Accident Rates - The accident rate for trucks is taken as 2.5×10^{-6} accidents per mile; for rail, 1.5×10^{-6} accidents per railcar mile, and for barges, 9.4×10^{-6} accidents per barge mile, based on available data.[10,11] An analysis of accident rates as a function of accident severity is contained in Appendix B of WASH-1238[1] and a more recent, more detailed analysis is given in "Severities of Transportation Accidents"[10] for all modes but barge transport.

4.9.4 Environmental Impacts

Weight and Traffic Density - There can be as many as 18 waste shipments by truck on public highways for the annual waste requirements of facilities supporting a model LWR, but excluding waste from the LWR itself (except for spent fuel in the no-recycle case). The TRU shipments may involve return of reusable shipping packages. According to the Federal Highway Administration, the average number of trucks per day on any section of the U.S. highway system varies from about 100 to 10,000.[12] Total truck miles traveled on U.S. highways in 1974 were estimated to exceed 55 billion.[13] The truck mileage associated with waste shipments related to a model LWR is, at most, 18,000 (Table 4.34), which is less than one millionth of total truck travel in 1974, and thus is too small to have a measurable effect on the environment from the increase in traffic density. The same conclusions hold true for rail shipments.

The number of drums of waste per vehicle can be adjusted so that the truck can stay within weight restrictions imposed on highway vehicles and railcars. Therefore, there need be no excessive loads on roadbeds or bridges.

Injuries, Fuel Use, and Fuel Emissions - The nonradiological environmental effects of the shipment of materials from the nuclear fuel cycle are similar to those characteristic of the trucking industry in general, in terms of injuries, fatalities, fuel use, and emissions. Fuel cycle waste transportation adds about 18,000 miles of truck travel, including return of empty casks and protective overpacks. According to the American Trucking Association, an intercity truck averages 4.9 miles per gallon of diesel fuel, and, during 1970, trucks consumed more than 25 billion gallons of diesel fuel. The 3700 gallons of fuel that would be used to transport nuclear waste in support of a 1000-MWe nuclear reactor is less than 10^{-6} of the fuel used by the trucking industry in 1970. Based on emission yields for diesel engines of 102, 16.8, 168, 12.3, and 5.9 kg per 1000 gallons of diesel fuel respectively for CO, hydrocarbons, NO_x, SO_x, and particulates, the combustion of 3700 gallons of diesel fuel would release about 0.38, 0.062, 0.62, 0.045, and 0.022 MT respectively, which are very small annual emissions.*

Using rates of 0.03 fatality and 0.51 injury per accident[14] yields 1.3×10^{-3} fatality per reactor year (about one death per 740 reactor years), and 2.3×10^{-2} injury per reactor year (about one injury per 40 reactor years) as transportation risks from common causes.

*"Final Environmental Statement, LWBR Program," ERDA-1541, June 1976, Table IX, G(A)-3.

Thermal Emissions[1,15] - Low-level waste shipments would have no readily detectable heat output. Shipments of filtration and evaporation residues will generate 0.4 Btu/hr of heat per package, which will have a negligible environmental effect.

Transuranic wastes, even those consisting of fuel-element hulls, generate less than 250 Btu/hr for typical shipments. This quantity of heat has no significant impact on the environment.

Typical HLW shipments will generate heat at the rate of 150,000 Btu/hr for waste generated at reprocessing plants, and at the rate of 330,000 Btu/hr when fuel assemblies are shipped from nuclear reactor sites. This heat generation can be compared with the 180,000 Btu/hr released by a 100-hp truck engine operating at full power. Because of the limited number of HLW shipments compared to truck traffic and other thermal sources, the thermal environmental impact is negligible.

The radioactive contents of HLW in the recycle option will add about 14 million Btu per reactor year to the environment during transportation.

Shipments of Irradiated Nuclear Fuel - Shipments of irradiated nuclear fuel from nuclear reactors (for the no-recycle case) to an interim storage facility, and then on to permanent disposal, have the same environmental impact as such shipments from nuclear reactors to FRPs, with suitable adjustment for differences in miles transported. Impacts associated with the latter shipments are assessed in WASH-1238,[1] and are incorporated in this assessment by reference. Impacts associated with irradiated nuclear fuel shipments to interim storage and disposal may vary by up to a factor of two from those described in WASH-1238 because of differences in miles transported.

Radiological Impact - Normal Transportation - Using the estimates and assumptions in the following paragraphs, the population exposures are calculated and summarized in Table 4.35.

Both truck and rail shipments are conservatively assumed to cover, on the average, 200 miles per day. A 500-mile truck trip would require 2-1/2 days and require two drivers. The drivers would spend 20 hours in the truck cab and one hour outside the truck at an average distance of 3 feet from the cargo for the 500-mile trip. The 1500-mile shipments are expected to be completed in 7-1/2 days, with drivers spending a total of 60 hours in the truck cab, and a total of 3 hours at an average distance of 3 feet from the cargo.

Garagemen or train brakemen might be expected to spend from one to ten minutes each in the vicinity of a railcar or the cab of a truck, for an average exposure of about 0.5 mrem each per shipment. For a 500-mile trip 5 garagemen (truck travel) or 5 brakemen (train travel) are expected to be thus exposed.[16]

Fuel-cycle wastes are transported as "full loads" on exclusive-use vehicles. Since the packages are not handled enroute, there would be no exposure of freight handlers, either by truck or by rail.

Table 4.35

RADIOLOGICAL IMPACTS FROM WASTE SHIPMENTS (NORMAL OPERATION)

Type of Shipment	Fuel Cycle Option: U Recycle Only — Exposure (Person-rem)	U Recycle Only — Number of People Exposed	Throwaway (no recycle) — Exposure (Person-rem)	Throwaway (no recycle) — Number of People Exposed
Low-Level				
Transport Workers	0.61	75	0.56	70
General Public - onlookers	0.14	70	0.13	65
- residents	0.13	2.2×10^6	0.12	2.0×10^6
TRU				
Transport Workers	1.25	80	-	-
General Public - onlookers	.12	80	-	-
- residents	.11	1.8 x 106		
			-	-
High-Level				
Transport Workers	.011	20	-	-
General Public - onlookers	0.046	20	-	-
- residents	0.041	6.8 x 105		
			-	-
Subtotals				
Transport Workers	1.87	175	0.56	70
General Public	0.59	4.7×10^6	0.25	2.0×10^6
TOTALS	2.46		0.81	

Members of the general public might be exposed to radiation from waste shipments at truck stops or rail stations while the vehicle is stopped. Although truck and rail shipments will be placarded "radioactive," it is assumed that 5 persons per day would be exposed for 3 minutes at 3 feet from the cargo.

Along a 500-mile truck or rail route, it is estimated that 150,000 persons reside within a distance of 0.5 mile from the transporting vehicle. Their doses are calculated for persons in an area between 100 feet and 0.5 mile on either side of the shipping route, assuming a population density of 127 persons per square kilometer (330 persons per square mile), a high average value for population density across the United States. Details of the dose calculation may be found in Appendix D of WASH-1238.[1]

Since fuel-cycle wastes are transported as full loads on exclusive-use vehicles, no significant exposure of transported animals is expected.

Radiological Impact - Accident Risks

Low-Level Wastes - In the shipment of packages of solid waste, it is estimated that about one in 10,000[17] packages may not be properly closed. If an improperly closed package comes open, the solid form of the material, either as compacted soft wastes or consolidated solid wastes, reduces the likelihood that radioactive material would be released or that significant radiation exposures would result. Because of the solid, low-activity radioactive contents, cleanup volumes for this type of incident would be small.

Based on a rate of 2.5×10^{-6} accidents per mile[10] and an average shipment distance of 500 miles, a truck shipment of low-level wastes might be involved in an accident once in about 800 shipments, or about once every 55 years. In such an event, the low specific activity and radiation levels of the materials limit the radiological impact to negligible levels.[18] Most of the radioactivity is bound in the waste. Unless fire ensues, the amount of radioactivity that would become airborne if a drum or package were to be broken open would not exceed a very small fraction of its contents. In a fire, combustible wastes may be burned, but most of the radioactivity in the burned waste will remain in the ashes.[19,20]

TRU Wastes - The radiation risk to the environment due to TRU wastes being involved in an accident is small; however, cleanup following a release to the environment could be expensive. A contamination incident in 1963 involved the spread of TRU waste in a truck terminal. The waste in that incident was in liquid form, causing cleanup costs of about $30,000.[21]

The same considerations hold true for low-level TRU waste under accident conditions as presented for low-level non-TRU waste. There is little chance of the low-specific-activity, low-radiation-level materials presenting a significant hazard even if consumed by fire.

Cladding hulls, however, constituting about half of the TRU waste, are highly radioactive, and must be packaged in accident-resistant containers to avoid dispersal of the radioactive contents in the event of an accident involving fire. The consequences of a fire which results in burning of cladding hulls are described in Section 4.3.3.

High-Level Wastes - The structural and containment features of casks for transporting HLW will be similar to those of casks for irradiated nuclear fuel as described in WASH-1238.[1] Furthermore, HLW will be packaged in completely sealed steel canisters that are in turn enclosed in a shipping cask so that two levels of containment will be provided. Conclusions with respect to the ability of irradiated fuel casks to withstand unusual accidents will thus be applicable to HLW casks.

For the 1500-mile shipment considered and an accident rate of 1.5 accidents per million railcar miles,[10] a shipment of HLW might be involved in an accident once in 440 shipments or once in 190 years. HLW containers must be designed and constructed to withstand accidents likely to be encountered in transport. The HLW is a stable material, sealed in leak-tight canisters. Thus, the cask design and waste form protect against significant releases of radioactivity. If an accident caused loss of cooling, the packages would get hot, but not sufficiently so to cause failure. A breach of shielding or containment could result in high radiation exposure in the vicinity of the accident.[22] Should any of the solid waste spill from a container, it would need to be removed by remote handling techniques.

Conclusion - Considering the low probability of a shipment of fuel cycle waste being involved in an accident, the requirements for package design and quality assurance, the nature and form of the fuel cycle waste, and the controls exercised over the shipment during transport, it is concluded that the radiation risk to the environment from fuel cycle waste in transportation accidents is small compared to the risk associated with radiation exposures in routine transportation of wastes.

4.10 Sabotage

Sabotage in the context of the nuclear industry means any deliberate act that could directly or indirectly endanger public health and safety by exposure to radiation.[1] The areas against which sabotage might be directed include all fuel-cycle operations (including reprocessing plants and fuel fabrication facilities), as well as interim storage facilities, disposal facilities, shipping packages, and transportation. Each of these cases is discussed below.

Even though the existence of motivation to sabotage nuclear facilities has not been proven, prudence has dictated that facilities and packages be protected against such acts and their possible consequences. Features designed into plants and packages to prevent releases or serious consequences due to accident or natural phenomena also provide protection against sabotage. In addition, certain preventive measures have been taken specifically to deter sabotage or to mitigate the seriousness of releases caused by sabotage.

The effectiveness of these measures cannot be easily tested since only if they fail will the value of deterrence be documentable. Designs of facilities and security systems are tested both physically and analytically, and these provide the major protection on which we rely to minimize the consequences of an act of sabotage. The testing of these measures gives some assurance that the risk from sabotage is small, but evaluation in the literature of the range of possible consequences is not complete.

Evaluation of risk of radiation release due to acts of sabotage is based upon consideration jointly of the probability of such an act, the probabilities of the sequence of events leading from the act to the release of radiation, and the consequences. The probability of the perpetration of an act of sabotage is governed by the motivation and capability of the saboteur. The probability of subsequent events depends on the design features (particularly the mitigating features) of the plant or shipping package, as well as on operational safeguards. The consequences of these events depend on the nature of pathways for transport of the nuclides to man.

In the public interest, NRC regulations provide that information regarding site-specific protection plans and countermeasures against sabotage can be withheld from public disclosure. However, it is possible to discuss in a general way the motivations and capabilities of saboteurs, protection requirements of the regulations, intrinsic features of system design that protect against sabotage, and the consequences of acts of sabotage, and this is done below.

4.10.1 Nature of the Threat of Sabotage

Attempts to define threats to the nuclear industry are based upon extrapolation from historical data dealing with violence in more conventional settings and/or aimed at more conventional materials. Threats to nuclear facilities can be aimed at either the release of radioisotopes or the diversion of fissionable material (in the present discussion, waste plutonium) for illicit use. Extrapolations from conventional forms

of violence to acts of nuclear sabotage are tenuous because no certain rationale has been demonstrated for a continuity between historical forms of violence and violence in the nuclear industry, particularly toward the acquistion and use of nuclear weapons.

A number of adversary groups have been identified historically as likely perpertrators of acts of sabotage. The adversary groups assumed to give greatest cause for concern are organized crime, terrorists, or dissidents.[2-4] The latter include political extremists and disgruntled employees in the nuclear power industry. Terrorist groups have become highly visible during the past decade,[5-10] and the potential threat of these groups to the nuclear power industry has been discussed.[2-4,7-9] The capabilities of these groups are often described in terms of their accomplishments, as well as in terms of what is known about their manpower resources, nationalities, political affiliations, funding, armaments, motives, dedication, and other attributes.

Speculations concerning motivations of adversary groups for threatening public health and safety by attacking nuclear fuel cycle facilities range through revenge, blackmail, extortion, ransom, publicity, and intent to create mass destruction.

Organized crime could nominally be considered as a potential adversary group because of its experience in direction and management of criminal activities. Arguments may be made, however, that in spite of its talents, organized crime would not be likely to enter into activities which safeguards are designed to protect.[11-17] Students of organized crime hold the opinion that fear of public and government reprisals would likely prevent any involvement, even the indirect one of black market operations in SNM, in which illicit nuclear activities could adversely affect the public. The risk of such ventures is high and the payoff small compared to other illegal activities.

Analyses of terrorism found in the popular press often state the concern that terrorists will acquire a nuclear capability and then be in a position to wield extraordinary powers of extortion or political blackmail.[18-22] Similar sentiments have been expressed in various monographs and terrorist studies.[7, 23, 24] Recently, however, terrorist motivations have been reassessed and earlier prevalent assumptions that the acquisition of means for inflicting mass destruction would be a natural evolution of terrorism no longer possess the currency they once had.[25-28]

Under appropriate circumstances, terror has proven to be both an effective and efficient psychological weapon. No other technique is as immediately available or offers as much return for relatively small investments as does selectively applied terror. Conditions are important, however, and

> ... the competent practitioners of terrorism usually know how their actions will affect their enemies and what reactions they can expect from those not directly involved.[29]

Implied in this quotation is the requirement that any terror campaign be limited. Most authorities agree that terrorism is generally a tactic of the weak. For terror

to be an effective coercive tactic, terrorists must be able to make the public understand what is being attempted, that penalties are involved, and that innocents will be spared to the degree possible.[30] Indiscriminate use of weapons of mass destruction, such as clandestine nuclear devices, would violate these criteria.

Historians and social scientists point out that many terrorist organizations with the capability of engaging in transnational terrorism are heavily subsidized by governments; some are, in fact, governmental branches. Since employment of weapons of mass destruction could precipitate countermeasures of such severity as to endanger governments associated with the act, such groups might be cautious about using them.

Indigenous terrorists would be constrained from using weapons of mass destruction, since they operate within their own countries and must retain favorable public opinion to satisfy their objectives. Recent history shows several instances where conventional indigenous terrorist violence was curbed.[6,26,31]

In addition to threat scenarios involving diversion of SNM for purposes of constructing a clandestine nuclear device, or sabotage of a nuclear facility to endanger local populations, other threat scenarios involving nuclear hoaxes, sabotage of a nuclear facility in the construction stage, seizure of a nuclear facility, radioactive contamination of a symbolic target, and dispersal of diverted plutonium may be considered. A list of such threat scenarios may be derived from Jenkins.[25] At domestic nuclear facilities, hoaxes, threats of violence, actual and attempted break-ins, and arson have occurred, but bombings, seizure, and contamination of a symbolic target have not.

Purposeful dispersal of plutonium does not appear to have any utility for a terrorist. The majority of effects on individuals exposed to plutonium inhalation are likely to appear fifteen or more years after exposure;[32] immediately measurable radiobiological effects are not likely. Thus plutonium dispersal lacks the threat ingredients important to terrorism--immediate effects and an obvious cause-effect relationship. However, a general threat of dispersal could nevertheless precipitate public fear.

No threat to the nuclear power industry of the types discussed above has been identified. No identifiable group with motivations to establish such a threat has been detected, nor has such a domestic threat occurred.

4.10.2 Protection of Fuel Cycle Plants and Materials Against Sabotage

Protection is achieved through regulations and through intrinsic features of system design. Each of these aspects is addressed below.

4.10.2.1 The Regulatory Approach

Even without a specific definition of a potential adversary, regulations have been implemented to prescribe a range of physical security measures that a licensee

must follow. The intent of present regulations is for the licensee to safeguard against the acts of a single insider and attacks by a small armed group.

For contingency planning purposes, several formal and informal information exchanges with other federal agencies concerned with security and intelligence relating to safeguarding domestic interests have been established by NRC. The NRC is also developing criteria for licensee contingency plans.

Safeguards Requirements for Wastes

The levels of protection provided for certain nuclear wastes against acts of industrial sabotage at FRPs and fuel fabrication plants are specified in Section 73.50 of 10 CFR Part 73 and are summarized in Annex 4.10.A. Principal features include protection forces (guards), physical and procedural access controls, detection aids, communication systems, and liaison with local law enforcement agencies. Any equipment, system, device, or material of which the failure, destruction, or release could directly or indirectly endanger the public health and safety by exposure to radiation is considered "vital," and is subject to protective measures set forth in 10 CFR Part 73.50. The site-specific identification of "vital equipment and material" is a necessary part of the NRC staff's review of the physical protection plan submitted by an applicant. Radioactive wastes considered to be "vital" are located in areas which are protected by two barriers and to which access is controlled.

Radioactive wastes at nuclear reactor sites are protected against industrial sabotage in accordance with criteria specified in Regulatory Guide 1.17 and American National Standards Institute Standard N18.17-1973. A proposed regulation, 10 CFR Part 73.55, presently under consideration by the Commission, would significantly upgrade physical protection requirements for nuclear power reactors. If adopted by the NRC, the planned power reactor regulations would closely follow, but in some cases would be more stringent than, requirements presently in effect for the protection of fuel cycle facilities processing SNM. The fuel cycle regulations are now being reviewed by the staff with the purpose of assessing their adequacy. Consequently, the level of protection afforded nuclear wastes before preparation for shipment would likely be increased. The adequacy of fuel-cycle regulations is now being assessed by the NRC staff.

No reprocessing plants are now operational. Of the two existing plants, the Nuclear Fuel Services (NFS) facility at West Valley, New York, is shut down for modification, and the Allied-General Nuclear Services (AGNS) facility at Barnwell, South Carolina, is under construction. The AGNS facility contemplates use of a process in which HLW will be stored as a liquid for a period of time. The NFS facility possesses 600,000 gallons of liquid waste in a storage tank (see Sec. 4.2.1). Future reprocessing plants will probably produce a solid form of HLW with only nominal surge tanks.

Liquid HLW at the AGNS facility is stored in vital areas additionally protected by the safety-related structural design. The cooling equipment for liquid HLW

storage tanks is also protected as vital. Additional safeguards requirements for proposed HLW solidification facilities and the associated solid products have not been determined, but are covered by safeguards requirements for the reprocessing plant itself.

Safeguards requirements are not imposed by regulations on other-than-HLW disposed of at commercial burial grounds; access controls imposed for health and safety reasons provide protection (see Sec. 4.7.3). Wastes at federal repositories will be subject to physical protection requirements imposed by NRC.

Safeguards Requirements for Spent Fuel

Until it is reprocessed to recover the remaining fissile material, spent fuel must be stored in accordance with requirements for its protection against sabotage given in 10 CFR Part 73 (see Annex 4.10.A) for FRPs and spent-fuel storage sites.

Spent fuel at reactor sites is subject to the same protection as other vital equipment at the reactor. Requirements for physical security at reactors are contained in 10 CFR Parts 50.34(c), 73.40, and 73.50. Regulatory Guide 1.17 and ANSI Standard 18.17 describe physical security criteria generally acceptable for protection of nuclear power reactors against acts of industrial sabotage.

A proposed regulation, 10 CFR Part 73.55, presently under consideration by the Commission, would codify physical security requirements at reactor power plants. The new requirements are comparable to those presently described in Part 73 for fuel-cycle facilities.

Unreprocessed spent fuel in storage at a reprocessing facility is subject to the physical protection requirements of 10 CFR Part 73 as prescribed for fixed-site storage. Unreprocessed spent fuel that has an external radiation dose rate in excess of 100 rems per hour 3 feet from any accessible surface without intervening shielding is exempt from the physical protection requirements of 10 CFR Part 73 while in transit because the high radiation levels and the heavy shielded casks required are viewed as adequate protection from sabotage.

4.10.2.2 Intrinsic Protection

Designs required to protect against radioactive releases due to rare and severe natural phenomena also protect against releases resulting from sabotage.

Fixed-Site Facilities

The portions of plants used for storing or processing plutonium must be designed to withstand the effects of natural phenomena:

- Penetration resistance to a spectrum of tornado-driven missiles
- Protection against tornado differential pressures of 3 pounds per square inch in 1.5 seconds

- Damage resistance against earthquake forces

- Reinforcements and waterproofing to protect against severe floods.[33]

Additional protection against sabotage stems from the massive biological shielding included in nuclear power plants, FRPs, and MOX fuel fabrication plants to protect workers from radiation hazard. This shielding may include underground construction and/or walls of up to 6 feet of reinforced concrete. These and other measures result in structures that provide a high level of inherent protection against explosions or other acts of sabotage.

A recent study of the vulnerability of nuclear power reactors to sabotage is instructive. The report describes:

"...characteristics of commercial nuclear power plants which greatly increase the difficulty of releasing radioactivity by sabotage:

- The "defense-in-depth" concept of reactor plant design;

- The massive structure of the plant, which protects critical components from external attack;

- The safety design basis of the plant, which emphasizes system reliability, flexibility, redundancy, and protection against common mode failures; and

- Engineered safety features, which are added to the basic system to cope with abnormal operations or accidents."[10]

Systems that help to prevent accidents or sabotage in fuel cycle facilities have been described in considerable detail.[34] Some of these systems, described below, also limit the consequences of either accidents or sabotage.

At separation facilities, PuO_2 conversion facilities, MOX-FFPs, and spent fuel or waste storage facilities, operations that have a potential for equipment failures or accidents which could disperse radioactive contaminants are performed within process cells or buildings. These structures are designed to maintain adequate confinement in the event of accidents or violent natural phenomena. It is expected that during the life of these facilities, some equipment failures will occur.

Accordingly, monitors are provided to detect such failures, and to provide or signal a need for corrective action. Moreover, these facilities are designed so they can be decontaminated for the repair or replacement of equipment. A ventilation system routes contaminated air from inadvertent releases through HEPA filters that remove airborne radioactive particulates. In summary, separation facilities and PuO_2 conversion facilities are designed to assure adequate margins of safety in accidents and to mitigate their consequences.

In fuel fabrication plants, the worst consequences stem from fires or explosions. Hence, possibility of these events is considered in detail in plant design, construction, and operation. Regulatory Guide 3.16 presents methods acceptable to the regulatory staff for a fire-protection program which should prevent, detect, extinguish, limit, or control fires and explosions and their related hazards. Licensees must operate within these or equivalent constraints.

Only a few accidents involving radioactive materials have occurred in existing facilities; none has resulted in significant contamination beyond the immediate vicinity of the plant, and the experience gained has resulted in improved safety procedures and features. To the extent that acts of sabotage initiate sequences of events much like those initiated by accidents, the measures designed into fuel-cycle facilities for mitigation of consequences of accidents also provide protection against the consequences of sabotage.

Shipping Packages

The measures taken to protect from radioactive releases in severe transportation accidents also protect from releases due to sabotage of shipping packages. Regulations require that packages be designed to performance standards with respect to shielding, containment, heat transfer, and nuclear criticality.[35] The more radioactive the contents, the heavier the package must be, and as a general rule, the heavier the package, the more difficult it is to damage it with high explosive. The weight of the package is thus deemed to deter sabotage as well as to mitigate its effect.

Intentional opening of spent-fuel shipping casks would require an appreciable amount of time, elaborate planning, and shielding and handling facilities.[36] Cask covers cannot be removed by hand because of their bulk and weight. Overhead cranes would have to be employed and the removal would have to be performed remotely, usually underwater, because of the high radiation field of the open cask. Even with such facilities a significant amount of time is required to remove a cover.[36] Opening either a rail or truck cask would require that the cask be off the vehicle in the vertical position. The tight packing of fuel elements in the cask, the difficulty of removing the cover, and the level of radioactivity of the exposed fuel protect against any introduction of explosives into the cask with the intent of propelling the fuel out of it.

Economic and engineering considerations dictate that shipping packages be passive systems, requiring minimum handling in the transportation system. These packages are usually handled by people at nodes in the transportation network but not on the links. Sabotage of nodal operation is protected against by plant security features: operations are executed by authorized personnel in fenced, guarded work areas that are monitored and alarmed. Protection against the effects of sabotage in link movements is furnished by physical barriers provided by the vehicle and packaging. Depending on the radioactivity of its contents, a package has sufficiently thick walls and mass to resist the dispersing effects of munitions and explosives that might be used by a saboteur. By DOT regulation, packages of radioactive material must be secured against movement within the vehicle. This requirement deters theft by making it difficult. For some spent-fuel package designs, a screen houses the cask itself and also serves to deter sabotage.

4.10.3 Environmental Effects of Sabotage

Radiologically, some sabotage events are similar to accidents or abnormal operations. Estimation techniques developed for the effects of these latter causes also apply to some sabotage events.

Acts which would result in widespread dispersal of radioactivity seem limited to the use of massive explosive force or the causing of an accident or fire as severe as the worst ones contemplated by the Reactor Safety Study.[37] With explosives, the large quantities required and the freedom of action necessary for a saboteur to engineer their placement and detonation argue against the likelihood of success. The destruction of nuclear facilities by fire does not appear to be an attractive option. The Rocky Flats Plant fire in 1969 caused no detected offsite release of toxicity and no deaths among the public have been attributed to the incident, even though there was extensive facility damage and burning of plutonium metal.

Recent consideration of acts of sabotage directed against current-generation LWRs is instructive regarding sabotage of fuel-cycle facilities. A current study[10] states:

> "Many factors influence the consequences: the sabotage option chosen, the operating status of the engineered safety features, the containment failure mode, the time and space variation of the wind and meteorological conditions, the site population distribution, and the extent of emergency response of on-site and off-site personnel. Control of all these factors is well beyond the capabilities of a credible sabotage operation. Accordingly, evaluation of the consequences arising from the sequences developed by the adversary teams yielded values that are a small fraction of the maximum consequences considered by the Reactor Safety Study."

Similar arguments pertain to waste and reprocessing facilities, though no comparable study has been made.

The incidents with the greatest potential for releasing radioactivity into the environment are those that cause melting of fuel. The most serious of these incidents would be a loss of coolant leading to melting of spent-fuel storage units. To the extent

that this is similar to core meltdown in a reactor, it is instructive to consider the latter. The most likely consequences of a core-melt accident are projected as follows:[37]

Type of Consequence:	Magnitude of Consequence:
Fatalities	Less than one
Injuries	Less than one
Latent fatalities per year	Less than one
Thyroid nodules per year	Less than one
Genetic defects per year	Less than one
Damage to off-site property	Less than $1,000,000

The Reactor Safety Study describes "Large Consequence Accidents," which would also include acts of sabotage, as follows:

> "Potential core melt accidents, occurring under typical or average values of radioactive release, weather, and exposed population, would have modest consequences. The reasons that probabilities are much smaller for large consequence events is that all the factors affecting consequences must be at or near their worst condition. Thus, they require a core melt accident coupled with additional failures that cause large radioactive releases coupled with unfavorable weather conditions and a very high population density exposed to the released radioactivity. Since the accident, the population, and the weather are generally independent, large consequence events are quite unlikely."[37]

Because of the decay of short-half-life fission products in the spent fuel, thermal energy from spent fuel at a reactor, reprocessing plant, or interim storage facility is less than for a reactor core. Therefore an act of sabotage to a spent-fuel facility is expected to result in less release than that calculated for a reactor core.

Spent Fuel Storage Pools

Sabotage of a spent-fuel storage facility has been analyzed for the Morris, Illinois, site.[36] Events considered include loss of water through sabotage of basin equipment, an explosion in or around the basin, a criticality event produced by sabotage, and sabotage of cask-contained fuel. Because there is no time-lag between attack and release, the event with perhaps the greatest potential for releasing radioactive material would be explosive rupture of the fuel rods. For a correctly placed charge of reasonable size, the study indicates that about 10 percent of a single fuel element would be in the volume of serious blast damage. A long series of events must then occur with correct timing for radioactive material to appear at the site boundary. The explosive charge must be placed correctly, the fuel cladding must be ruptured, fuel material must be fragmented extensively, the fuel particles must be transported through the pool water to the air, and the air-suspended particles must pass out of the building and remain airborne to the site boundary. Considering that particle aggregation, precipitation, water entrapment, filter retention, and surface deposition all act to reduce the quantity of airborne material, the probability of a

radioactive aerosol appearing at the site boundary, according to this analysis, is quite small.

In one case, assuming that 10 percent of the affected rods are ruptured and that 0.1 percent (a very high percentage) of the fuel is pulverized into particles less than 10 microns in size and that the resulting fuel aerosol escapes to reach the site boundary, the whole body dose is calculated to be 20 mrem.

High-Level Liquid Waste Storage Tanks

The protections afforded by security systems at the reprocessor must be penetrated by a saboteur party to even be able to create situations that could lead to radiological release. Assuming that the security systems could be penetrated, the consequences of a plausible sequence of events in which the liquid HLW in a storage tank is caused to boil and escape through the pipe tunnels, equipment buildings, and filter stations have been calculated.[38] Suspended particles and vapors are assumed to be released to the atmosphere at ground level, without regard to the mitigating effects of filtration barriers normally present, and without consideration of any corrective action that might occur during the long time period (about 5 days) required for release of radioactive material.

The radiological consequences of such a release are evaluated by comparing the calculations of whole body doses and doses to various organs to established thresholds for observed health effects.[39] The probability of an individual within a unit area (unit population density) experiencing a health effect can be calculated as a function of distance from the release point. The total number of health effects depends on the population exposed to the release, which can be obtained by integrating the product of population density and area, both of which depend on distance from the release point, weighted by this probability.

For a uniform population density of about 10 people per square kilometer and a tank content of waste accumulated over half a year of reprocessing, the total number of effects are calculated by this procedure to be no early or latent deaths (see Ref. 30 for a fuller discussion, including other assumptions made).

Shipping Packages

Packages containing the greatest sources of radioactivity are casks for spent fuel and HLW. Releases from these casks are difficult to effect because the casks are massive and leaktight, and the contents are solid. Sabotage of packages of TRU and low-level non-TRU wastes would be much less effective if successful because of the lesser radioactivity of the contents. Consequently attention is focused on sabotage of spent-fuel and HLW casks.

Both spent fuel and HLW are intensely radioactive and generate considerable decay heat. Casks used to transport these materials are massive, thickly shielded structures designed to performance criteria[35] that represent accident conditions.

Design features that enable shipping casks to withstand severe transportation accidents (e.g., multiple heavy steel shells, a water jacket, and a thick, dense gamma shield) also enable them to withstand attack by small-arms fire and explosives. Extraordinary skills and materials would be required to breach the vessel.

A massive rupture of the cask is considered to be an incredible event, but a small-bore penetration into the inner vessel is conceivable. Calculations (assuming several values for the release fraction of radioactive contents) indicate that the consequences of such an action would be small for both spent-fuel and HLW rail casks.[40] For the largest release fraction considered, the number of effects in a population density of 100 people per square mile and averaged over many weather histories are calculated to be about 0.6 early deaths and 300 latent cancer fatalities in the spent fuel case and about 2 early deaths and 90 latent cancer fatilities in the HLW case (see Ref. 38 for a fuller discussion).

Interim Storage Facilities

For the storage method used in this report, (a rugged thick-walled structure placed outdoors and cooled by convective air flow; see Sec. 4.2.5), practically the only type of sabotage event that could result in radiological release would be the use of explosives. However, the introduction of explosives would be difficult and risky because of plant security. Even if uncommonly large amounts of explosive were introduced and detonated strategically when the waste was least protected (in transfer buildings), the chief effects would be contamination and destruction of the buildings. The amount of radioactive material appearing at the site boundary would be small. Health effects can be estimated by means of the sabotage consequence calculations discussed above.[38,40]

Burial Grounds

Sabotage of burial grounds leading to release is not likely. Among features protective against such sabotage are containment of the waste material in various kinds of packaging, underground burial, and immobilization of chemical waste in concrete or similar material.

Recently waste material was removed without authorization from the burial ground at Beatty, Nevada.[41] There were some parallels to sabotage. A large number of items were removed for sale or use and were later recovered; of those that were contaminated, the typical surface dose rate was 2 mr/hr. Items recovered from private houses were contaminated only with trace amounts. Consequently, both state and federal radiation specialists concluded that no health hazard existed as a result of these illicit acts.

Geological Repositories

The probability of human intrusion in a repository has been assessed as falling between 10^{-7} and 10^{-4} (Sec. 4.4.2). This activity implies not only sabotage but also mining, random wildcat drilling, and wars. The most vulnerable parts of repository operations appear to be those involved with transportation.[42] The consequences of transportation sabotage are discussed above; the consequences of sabotage of other parts of the repository are not expected to be significant in comparison.

ANNEX 4.10(A)

PHYSICAL PROTECTION REQUIREMENTS

Legal Basis

NRC responsibility for nuclear security derives from the Atomic Energy Act of 1954, as amended; and from the Energy Reorganization Act of 1974, which provides that "all licensing and related regulatory functions of the Atomic Energy Commission be transferred to the NRC. The Atomic Energy Act explicitly authorized the AEC to set standards and impose regulatory controls over nuclear materials in order to "promote the common defense and security or to protect health or to minimize danger to life or property."

The essentials of the safeguards system formulated by the AEC and now implemented by the NRC are found in regulatory requirements. Supplementary information appears in various Regulatory Guides issued to assist applicants in complying with these regulations.

Applicable Materials

The NRC regulations specifying safeguards requirements are written in 10 CFR Part 73. These regulations apply to strategic quantities of special nuclear material, defined as uranium-235 (contained in uranium enriched to 20 percent or more in the U-235 isotope), uranium-233, or plutonium alone or in any combination in a quantity of 5,000 grams or more computed by the formula, grams = (grams contained U-235) + 2.5 (grams U-233 + grams plutonium).

Summary of Safeguards Requirements at Fixed Sites

A physical protection plan must be submitted by each license applicant to the NRC for approval, based on compliance with the following features:

Physical Security Organization

The licensee must maintain a physical security organization, including armed guards, to protect his facility against industrial sabotage. At least one supervisor of the security organization must be onsite at all times. The licensee must establish, maintain, and follow written security procedures which document the structure of the security organization and which detail the duties of guards, watchmen, and other individuals responsible for security. All guards or watchmen must be properly trained, equipped, qualified, and requalified at least annually.

Physical Barriers

All "vital equipment", which is defined as any equipment, system, device, or material whose failure, destruction, or release could directly or indirectly endanger public health and safety, must be located within a separate structure or barrier designated as a "vital area." All vital areas must be located within a large protected area which is surrounded by a physical barrier. An isolation zone is required around the outer physical barrier and it must be kept clear of obstructions, illuminated, and monitored to detect the presence of individuals or vehicles attempting to gain entry to the protected area and to allow response by armed members of the facility security organization to suspicious activity or to the breaching of any physical barrier.

Access Controls

Personnel and vehicle access into a protected or vital area must be controlled. A picture badge identification system must be used and visitors must be registered and escorted. Individuals and packages entering the protected area are required to be searched. Admittance to a vital area must be controlled and access limited to persons who require such access to perform their duties. Keys, locks, combinations, and related equipment are required to be controlled to minimize the possibility of compromise.

Intrusion Alarms

All emergency exits in the protected area and vital areas must be alarmed. Each unoccupied vital area must be locked and alarmed. All alarms must annunciate in a continuously manned central alarm station located within the protected area and in at least one other continuously manned station. All alarms must be self-checking and tamper-indicating, and inspected and tested for operability and required functional performance at specified intervals not to exceed 7 days.

Communications

Each guard or watchman on duty must be capable of maintaining continuous communications with an individual in a continuously manned central alarm station within the protected area and who must be capable of calling for assistance from other guards and from local law enforcement authorities. To provide the capability of continuous communication with local law enforcement authorities, two-way radio voice communication must be available in addition to conventional telephone service. All communications equipment must remain operable from independent power sources in the event of loss of primary power, and must be tested for operability and performance at least once at the beginning of each security personnel work shift.

Response Capability

Licensees must establish liaison with local law enforcement authorities and be prepared to take immediate action to neutralize threats to the facility. Such action may mean appropriate direct action on the part of the licensee, a call by the licensee for assistance from local law enforcement authorities, or both.

Records

Security records must be maintained of all individuals authorized access to vital and material access areas, including visitors, vendors, and others not employed by the licensee. Routine security tours, and all of the tests, inspections, and maintenance on security-related equipment and structures must be documented. A record must be maintained on each alarm, false alarm, alarm check, intrusion indication, or other security incident, to include the details of response by facility guards.

Reports to NRC

Attempts or acts of industrial sabotage must be reported immediately to NRC, followed by a written detailed report within 15 days.

Summary of Safeguards Requirements in Transit

Each licensee who transports, or delivers to a carrier for transport, specified special nuclear materials must submit a plan to NRC for review and approval, outlining the methods to be used for the protection of the special nuclear material in transit.

General requirements are as follows: If a common or contract carrier is used, the SNM must be transported under the established procedures of the carrier which provide a system for the physical protection of valuable material in transit and require a hand-to-hand receipt at origin and destination and at all points en route where there is a transfer of custody. Transit times of all shipments must be minimized, and routes selected to avoid areas of natural disaster or civil disorders. Special nuclear material must be shipped in containers sealed by tamper-indicating-type seals. The outer container or vehicle must be locked and sealed. No container weighing 500 pounds or less may be shipped on open vehicles, such as open trucks or railway flatcars.

All shipments by road must be made without scheduled intermediate stops. All motor vehicles must be equipped with a radiotelephone. Calls to the licensee or his agent must be made at predetermined intervals, normally not to exceed 2 hours, and if calls are not received when planned, the licensee or his agent must immediately notify an appropriate law enforcement authority and the NRC. Shipments by road must be accompanied by at least two people in the transport vehicle. When a specially designed transport vehicle with immobilization and penetration-resistant features is used, armed guards are not required. In the absence of immobilization features, armed guards must accompany the shipment. In instances when the transport vehicle has neither immobilization nor penetration-resistant features, at least two armed guards must accompany the shipment in a separate escort vehicle equipped with a radiotelephone.

Air shipments of SNM in quantities exceeding 20 grams or 20 curies, whichever is less, of plutonium, ^{233}U, and in excess of 350 grams of ^{235}U (enriched to 20% or more in the ^{235}U isotope) are prohibited on passenger aircraft. Shipments on cargo

aircraft are required to be arranged so as to minimize the number of scheduled transfers. Such transfer, when necessary, must be monitored by armed guards.

Rail shipments must be escorted by two armed guards in the shipment car or in an escort car. Continuous on-board radiotelephone communications capability must be provided with conventional telephone backup. Periodic calls to the licensee or his agent are required at the same time intervals as for road shipments.

Sea shipments must be made on vessels which make a minimum number of ports of call. Transfer at domestic ports from other modes of transportation must be monitored by a guard. Shipments must be placed in a secure compartment which is locked and sealed. Export shipments must be escorted by an authorized individual, who may be a crew member, from the last port in the U.S. until they are unloaded in a foreign port. Ship-to-shore communications must be made every 24 hours to relay position information and the status of the shipment as determined by daily inspection.

Reports on nuclear shipments are also required. A licensee who makes a shipment must notify the consignee of the shipment schedule and details, including its estimated time of arrival. In addition, the licensee is required to notify the NRC Regional Office of the shipment schedule days in advance of shipment. A licensee who receives a shipment must immediately notify the shipper. Shipments which fail to arrive at the destination on time must be traced. Unaccounted-for shipments must be reported immediately to NRC, followed by a detailed written report within 15 days.

4.11 Other Considerations

In addition to the foregoing discussions of environmental impacts we have also considered socioeconomic effects and effects on biota other than man.

4.11.1 Socioeconomic Effects

While socioeconomic impacts will be caused by establishment of reprocessing plants and waste-related facilities, the types of impact are not expected to differ in quantity or quality from those associated with any commercial nuclear power plant. Confirmatory assessment studies of operating nuclear facilities performed by the staff and other researchers, as well as staff experience with community impact analyses in the environmental impact statement process, lead to the general conclusion that stresses tend to be short-term and that communities are generally able to accommodate socioeconomic impacts without undue difficulty.[1]

The character and magnitude of the impacts are site-specific and determined by the size of the labor force, the size of the local population, the number of incoming workers in relation to population size, the capacities of facilities impacted, the administrative capability of the impacted political jurisdictions, and other factors. In rural areas with low population and an inadequate labor pool, large construction labor-force requirements cause immigration, which imposes stresses on local housing and other facilities and services. Areas less remote from urban centers usually have an adequate labor pool, and impacts are of less consequence.

Impacts that can be expected are comparable to or less than those caused by LWR construction activities and could include noise and dust around the site; disruptions or dislocations of residences or businesses; physical or public-access impacts on historic, cultural, and natural features; impacts on public services such as education, utilities, the road system, recreation, public health, and safety; increased tax revenues in jurisdictions where facilities are located; increased local expenditures for services and materials, and social stresses.

While the magnitude of given impacts and the nature of proposed mitigative measures are site-specific, the following general information can be presented: anticipated sizes of construction and operating work forces for reprocessing plants and waste-related facilities; and an example of site-specific socioeconomic evaluation made by an FRP applicant.

Table 4.36 presents a comparison of anticipated peak construction work forces for different types of facilities. The construction force for an FRP, which is expected to range from 1300 to about 2000, is the largest force estimated for construction of back-end fuel-cycle facilities, and is comparable to that needed for two-unit nuclear power stations.[2] While the operating work forces for reprocessing and HLW repository facilities are larger than those required for commercial nuclear plants, they are comparable to those of a moderate-size manufacturing facility, and the impacts would vary with location.

Table 4.36

A COMPARISON OF WORKER REQUIREMENTS FOR REPROCESSING AND WASTE-RELATED FACILITIES

Type of Facility	Peak Construction Work Force	Operating Work Force
Reprocessing	1,150 - 2,046	300 - 540
High Level Waste Repository	---[a]	200b
High Level Waste Interim Storage Facility	250[c]	150[d]
Low Level Waste Burial Facility	---[e]	25
Waste Solidification Facility	1,500[f]	--[f]
Multiple Unit Commercial Nuclear Power Plants	2,200[g]	170[g]

[a] The specific size of a construction work force has not been estimated, but is judged to be between that required for construction of an HLW interim storage facility and that for a waste solidification facility.

[b] Average annual number of workers.[4]

[c] Initial peak construction activity.[5]

[d] Figure presented is for initial operation, and would increase to 390 in 25 years.[5]

[e] Specific number of persons not reported but considered to be small.

[f] The estimated anticipated construction work force is provided in U.S. Nuclear Regulatory Commission, Draft Supplement to the Final Environmental Statement, Barnwell Nuclear Fuel Plant, Docket No. 50-332, June 1976, Fig. IV-2, p. IV-3. Operating work force was not reported.

[g] The anticipated work forces in this example are provided in U.S. Atomic Energy Commission, Final Environmental Statement, Bellefonte Nuclear Plants, Units 1 and 2, Docket Nos. 50-438 and 50-439, June 1974, pp. 4-5 and 11-5.

A recent applicant's site-specific environmental analysis can be used to provide perspective on potential socioeconomic impacts associated with an FRP.[3] The environmental evaluation concludes that the proposed facility will create impacts in a four-county area during construction. The impacts will be primarily in housing, education, and transportation, with no significant impacts on municipal services being identified. The evaluation includes an analysis which suggests that impact costs will be offset by revenues, but because socioeconomic considerations are not easily quantified, ongoing liaison with local leaders is anticipated.

Using information from the Tennessee Valley Authority, it is estimated that 25 percent of 2,000 peak construction workers and 300 supporting workers would move into the area and that 70 percent of the movers would have families including one school-age child. The 600 workers would require 100 conventional homes, 300 mobile homes, and 200 apartments and rooms. Analysis of existing housing showed that the most significant local housing impact would occur in one county where additional mobile-home sites would be needed, and that septic-tank problems could occur if the density of such homes were not regulated. Using state classroom standards, the study determined that the ~400 added students would need 11 classrooms and 11 teachers. Transportation impacts were identified for one road where through normal growth and construction traffic, the peak-hour capacity of 1,100 vehicles per hour would increase to nearly 1,400 during peak construction, producing a slight deterioration in traffic conditions.

A net decline in in-movers is anticipated between the construction and operation phases. Compared to the 800 total in-movers in the construction phase, in the operating phase the report estimated that 225 permanent and induced workers would move into the area, requiring 114 conventional homes, 88 apartments and rooms, and five additional classrooms and teachers for the 90 added students.

The applicant's mitigating efforts will include continuing relationships with state and local officials and institutions to permit identification of socioeconomic impacts and allow the planning of remedial measures. Liaison will be maintained with school officials, state and local highway departments, local labor sources and training facilities, local government agencies, and private housing developers to provide information and planning consultation.

While community impacts in some evaluation reports for waste-related facilities have not been treated at the same level of detail, the sizes of the work forces relative to reprocessing-plant work forces would suggest that socioeconomic impacts should be manageable through proper planning and mitigative efforts.

Socioeconomic impacts associated with reprocessing plants are expected to be similar in kind and magnitude to those for multiple-unit commercial nuclear power plants. While the magnitude of these impacts varies by location, experience with nuclear power plants indicates that even potentially severe impacts are manageable. Each back-end fuel-cycle facility is expected to serve a number of LWR plants. If

the impacts of the facility are allocated over the LWR plants serviced, the allocation makes a relatively small contribution to the total socioeconomic impact associated with a large multiple-unit commercial nuclear power plant. In addition, reprocessing plants and most waste-related facilities are subject to analysis of socioeconomic impacts within the framework of an environmental impact statement. For these reasons socioeconomic impacts have not been included in Table 2.10.

4.11.2 Effects on Biota Other than Man

The NRC staff reviews nonradiological impacts on both terrestrial and aquatic biota before issuing construction permits or operating licenses for facilities. Most of the experience behind such assessments comes from LWR licensing; however, the staff has found that nonradiological impacts caused by construction of nuclear facilities are basically similar to those that might be expected from many other large industrial facilities.

Terrestrial Assessments

Possible biotic impacts that are routinely reviewed include preemption of land or habitat, chemical emissions, and direct destruction of organisms.[1]

In assessing impacts on terrestrial ecosystems the staff considers preemption of dedicated public lands such as wildlife preserves, parks, and recreation areas. It also considers valuable land such as prime farmland, unusually productive ecosystems, or critical habitats of threatened or endangered species.

The acceptability of land preemption depends on the amount of land involved, the quality and economic value of land resources or products, the degree of protection afforded by law, zoning ordinances or regional plans, projected future land needs, the local scarcity or abundance of similar lands, and available alternatives.

The principal method of protecting valuable biotic resources is through siting regulation. Final siting approval for nuclear facilities is always based on site-specific studies; however, land preemption of up to 30,000 acres at a single site has been approved in reactor licensing cases,[2] although a more likely requirement is from 1100 to 11,000 acres.[3]

Single waste-management or reprocessing sites are unlikely to have permanent land requirements exceeding a few thousand acres (Table 4.37). Siting of these facilities is therefore not likely to create land-use impacts that are outside the licensing experience of the staff. The staff's methodology for siting assessments as related to biotic parameters and land use is considered applicable for waste management and fuel reprocessing facilities.

Biological effects resulting from chemical emissions are considered in licensing assessments, though such emissions do not normally affect nuclear-facility siting since they are usually subject to varying degrees of control.

Emissions from reprocessing facilities may include fluoride and gaseous oxides from fuel combustion. Of these, fluorine or its compounds present the highest risks of damage to vegetation, where it can cause necrosis and yellowing of leaves. In animals, dental fluorosis is a possible result of fluoride-contaminated food chains.

Table 4.37

ACRES OF LAND COMMITTED TO FUEL REPROCESSING AND WASTE DISPOSAL FROM 1975 THROUGH 2000

REPROCESSING[a]			BURIAL GROUND			FEDERAL REPOSITORIES		
Land Use	Total Acres (6 plants)	Acres per Reactor[b]	Land Use	Total Acreage[c] (11 fac.)	Acres per Reactor[b]	Land Use	Total[d] Acres (2 fac.)	Acres Per Reactor[b]
Undisturbed	9,000	17.8				Buffer	4,022	7.9
Disturbed	990	2.0	Surface Facilities	1,100	2.2	Surface	500	1
Permanently Committed	120	0.2				Subsurface	915	1.8

[a]Acres based on 6 multiples of Barnwell Nuclear Fuel Plant.[4]

[b]Based on 507 reactors in year 2000. Table III-2 GESMO.[5]

[c]Based on need for 11 facilities in year 2000. Table III-2 GESMO.[5] One hundred acres each facility, all permanently committed.

[d]Based on need for 2 facilities in year 2000 with 1-mile-radius buffer zone each.[5] All land assumed permanently committed.

The staff establishes whether fluoride concentrations could reach toxic thresholds to biota and imposes mitigating requirements if needed. Such analyses have been made of fluorides and gaseous oxides for the Midwest Fuel Recovery Plant[6] and the Barnwell Nuclear Fuel Plant.[4] Fluoride removal equipment was recommended for the Midwest plant.[6]

Direct destruction of organisms takes place primarily during construction of nuclear facilities as a result of site clearing and earth-moving. Losses of wildlife and habitat are therefore unavoidable. The NRC assessment made before issuance of a construction permit includes evaluation of the magnitude of wildlife loss and permits procedures to be developed that will prevent unnecessary loss.

Direct destruction of organisms is a possible though rare effect of operating nuclear facilities. Examples include bird collisions with tall structures or guy wires. Reprocessing or waste management facilities are unlikely to include structures that could cause effects more serious or widespread than have already been encountered in licensed reactors.

Aquatic Assessments

Impacts on aquatic biota are caused primarily by facility cooling systems. Adverse effects may accrue from thermal or chemical discharges and from entrainment, impingement, or entrapment of organisms in water intake or discharge structures.

Regulatory guidance has been developed for assessment of such effects.[7] Potential impacts on fisheries may be considered either as a siting problem or an engineering design problem. The question as to which option shall be used to protect fisheries can be resolved on a cost/benefit basis.[5]

The staff has recommended however that if critical or exceptionally complex systems have to be studied in detail to determine appropriate plant designs, proposals to use such sites should be deferred unless sites with less complex characteristics are not available.[6] It has been suggested for the purpose of siting that the amount of water which can be diverted for plant use be in the range of 10%-20% of flow and that minimum zones of passage for migrating aquatic species range from 1/3 to 3/4 of the width or cross-sectional areas of streams.[5]

Water withdrawals from large water bodies on the order of 4,000 cfs have been assessed in reactor licensing cases.[2] In more usual cases where closed-cycle cooling is used, withdrawals of the order of 89 cfs have received licensing approval.[3] FRPs will use less than these amounts. The Barnwell plant for example will use about 6.4 cfs (2880 gpm) of well water and will not be dependent on surface water. Thermal discharges from the BNFP will be mitigated by the use of a holding pond and the actual discharge of plant water to a surface water body amounts to only about 5 cfs, which is less than has been approved in reactor licensing cases. The staff, therefore, concludes that FRPs will not cause nonradiological impacts to aquatic biota that are outside

staff assessment experience or outside the range of facilities that have received licensing approval in other cases.

Waste burial facilities and repositories have small service water requirements and will not be dependent on large volumes of surface water for heat rejection. Impacts on aquatic biota from these facilities are therefore likely to be considerably less than those which have received licensing approval in other cases. Review and regulation of even small discharges may be required, however, even though substantial impacts on biota are not expected.

Nonradiological impacts on biota associated with reprocessing plants or waste disposal facilities are expected to be similar in kind and magnitude to those applicable to multiple-unit commercial nuclear power plants. The magnitude of these impacts varies by location, but experience with nuclear power plants indicates that even potentially severe impacts will be manageable. Each reprocessing or waste disposal facility is expected to serve a number of LWR plants. When the impacts of these facilities are allocated over the LWR plants serviced, the incremental biotic impact resulting from fuel-cycle activities makes a relatively small contribution to the overall biotic impact of a multiple-unit commercial nuclear power plant. In addition, nonradiological biotic impacts of fuel-cycle facilities are considered in the environmental impact statements for these facilities. For these reasons nonradiological biotic impacts have not been included in Table 2.10.

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APPENDIX A

NUCLIDES OF INTEREST

Summary of Activities (Ci/MTU) for Radionuclides of Interest as a Function of Post-Irradiation Time

(NOTE: Nuclides whose contribution to the specific activity of the fuel is less than 0.001 Ci/MTU at a post-irradiation time of 150 days are left blank.)

Power = 30MW; Burnup = 33,000 MWd: Flux = 2.92×10^{13} n/cm^2-sec

Z	Radionuclide	Half-life*	Curies/Metric Ton Uranium			
			Discharge**	160 days**	1 year**	10 years**
1	H-3	12.26y	7.09E02	6.91E02	6.7E02	4.03E02
4	Be-10	2.7×10^6y	-	-	-	-
6	C-14	5770y	-	-	-	-
20	Ca-41	1.1×10^5y	-	-	-	-
27	Co-60	5.27y	-	-	-	-
28	Ni-59	8×10^4y	-	-	-	-
34	Se-79	7×10^4y	3.98E-01	3.98E-01	3.98E-01	3.98E-01
36	Kr-85	10.4y	1.13E04	1.10E04	1.06E04	5.96E03
37	Rb-87	4.7×10^{10}y	-	-	-	-
38	SR-89	50.4d	7.18E05	8.51E04	5.53E03	5.15E-16
	SR-90	28y	7.76E04	7.68E04	7.57E04	6.07E04
39	Y-90	64.2h	8.07E04	7.68E04	7.58E04	6.07E04
	Y-91	57.5d	9.38E05	1.43E05	1.28E04	1.89E-13
40	Zr-93	9.5×10^5y	1.89E00	1.89E00	1.89E00	1.89E00
	Zr-95	65d	1.37E06	2.49E05	2.80E04	1.67E-11
41	Nb-93m	3.7y	1.45E-01	1.83E-01	2.31E-01	8.40E-01
	Nb-95	35d	1.38E06	4.73E05	5.95E04	3.61E-11
42	Mo-93	10^4y	-	-	-	-
43	Tc-99	2.1×10^5y	1.43E01	1.43E01	1.43E01	1.43E01
44	Ru-103	40d	1.22E06	7.41E04	2.05E03	2.09E-22
	Ru-106	1.0y	5.45E05	4.03E05	2.73E05	5.50E02
46	Pd-107	7×10^6y	1.10E-01	1.10E-01	1.10E-01	1.10E-01
47	Ag-110m	249d	3.68E03	2.37E03	1.35E03	1.66E-01
50	Sn-126	10^5y	5.46E-01	5.46E-01	5.46E-01	5.46E-01
51	Sb-125	2.0y	8.7E03	7.89E03	6.83E03	6.78E02
52	Te-127m	105d	1.54E04	5.77E03	1.57E03	1.30E-06
	Te-129	67.3m	3.37E05	1.42E03	2.17E01	0.0
53	I-129	1.72×10^7y	3.71E-02	3.74E-02	3.74E-02	3.74E-02
	I-131	8.05d	8.61E05	9.23E-01	1.99E-08	0.0
55	Cs-134	2.19y	2.46E05	2.12E05	1.76E05	8.38E03
	Cs-135	2.0×10^6y	2.86E-01	2.86E-01	2.86E-01	2.86E-01
	Cs-137	30y	1.08E05	1.07E05	1.05E05	8.56E04

Z	Radionuclide	Half-life	Discharge	160 days	1 year	10 years
58	Ce-144	285d	1.11E06	7.52E05	4.56E05	1.51E02
61	Pm-147	2.5y	1.02E05	9.73E04	8.39E04	7.75E03
63	Eu-154	16y	6.99E03	6.86E03	6.69E03	4.53E03
	Eu-155	1.7y	7.48E03	6.33E03	5.11E03	1.63E02
82	Pb-210	21y				
84	Po-210	138d				
86	Rn-220	51.5s	1.50E-03	2.29E-03	3.40E-03	1.60E-02
	Rn-222	3.82d				
88	Ra-224	3.64d	1.50E-03	2.29E-03	3.40E-03	1.60E-02
	Ra-225	14.8d				
	Ra-226	1622y				
ACTINIDES						
90	Th-228	1.91y	1.49E-03	2.27E-03	3.38E-03	1.60E-02
	Th-229	7340y				
	Th-230	80y				
	Th-232	14y				
	Th-234	24.1d	3.14E-01	3.14E-01	3.14E-01	3.14E-01
91	Pa-226	1.8m				
92	U-233	1.62×10^5y				
	U-234	2.48×10^5y	7.51E-01	7.55E-01	7.59E-01	8.30E-01
	U-235	7.13×10^8y	1.71E-02	1.71E-02	1.71E-02	1.71E-02
	U-236	2.39×10^7y	2.88E-01	2.88E-01	2.88E-01	2.88E-01
	U-238	4.51×10^9y	3.14E-01	3.14E-01	3.14E-01	3.14E-01
TRU						
93	Np-237	2.2×10^6y	3.33E-01	3.40E-01	3.40E-01	3.43E-01
94	Pu-236	285y	3.50E-01	3.16E-01	2.75E-01	3.08E-02
	Pu-238	89y	2.72E03	2.82E03	2.86E03	2.70E03
	Pu-239	2.44×10^4y	3.18E02	3.23E02	3.23E02	3.23E02
	Pu-240	6.58×10^3y	4.77E02	4.77E02	4.77E02	4.79E02
	Pu-241	13y	1.05E05	1.03E05	1.00E05	6.53E04
	Pu-242	3.79×10^5y	1.38E00	1.38E00	1.38E00	1.38E00
	Pu-244	7.6×10^7y				
95	Am-241	458y	8.59E01	1.59E02	2.50E02	1.41E03
	Am-242	16h	6.34E04	9.14E00	9.12E00	8.75E00
	Am-243	7.95×10^3y	1.81E01	1.82E01	1.82E01	1.82E01
96	Cm-242	163d	3.34E04	1.70E04	7.12E03	7.18E00
	Cm-243	35y	3.71E00	3.68E00	3.68E00	2.99E00
	Cm-244	17.6y	2.44E03	2.40E03	2.35E03	1.67E03
	Cm-247	4×10^7y				

* Handbook of Chemistry and Physics, 45th Edition, Section B

** ORNL-4628, "ORIGEN-The ORNL Isotope Generation and Depletion Code", May 1973, Tables A-7 and A-9.

APPENDIX B

ERDA WASTE MANAGEMENT PROGRAMS

I. Introduction

The radioactive waste management responsibilities of ERDA may be grouped in three categories:

- Storage or permanent disposition of commercial wastes that may be identified by NRC regulations as requiring transfer to federal custody;

- Storage or permanent disposition of radioactive wastes generated at ERDA-operated sites. (Since most of these wastes are from national defense programs, they are frequently referred to as "defense wastes");

- Research and development activities, including development of waste treatment techniques to facilitate storage or disposition.

A detailed plan for ERDA's long-range waste-management goals appears in "A National Plan for Energy Research and Development; Creating Energy Choices for the Future," ERDA 76-1, Energy Research and Development Administration, WA, DC.

II. The ERDA Radioactive Waste Management Program

Each step in the nuclear fuel cycle generates radioactive waste. Present technology allows the treatment and packaging of waste into forms suitable for disposal. Liquid and gaseous effluents from these operations are controlled so that radiation levels near nuclear facilities are not only well within international standards but are very small fractions of the lowest natural radiation background dose that can be incurred by people residing in the U.S.

Most of the radioactivity in the nuclear fuel cycle is found in the wastes from the first cycle of the extraction process in the chemical processing of partly spent nuclear fuel. Commercial high-level radioactive waste, under present NRC regulations, must be delivered to federal custody in solid, stable form no more than 10 years after its generation. On a near-term basis, management requirements for this waste are containment, shielding against intense penetrating radiation, and dissipation of heat from radioactive decay. However, long after normal decay of fission products has eliminated needs for shielding and heat removal, continued isolation will be required because of a few materials (principally traces of unrecovered plutonium-239) which combine long radioactive half-life with high radiotoxicity per unit weight. Because Pu-239 and related materials (known collectively

as transuranic nuclides) will also be generated in the recycling of plutonium as a fuel, a proposed NRC regulation requires that transuranium (TRU)-contaminated commercial waste be transferred to federal custody in solid form no more than 5 years after its generation.

These wastes must be isolated from man's environment for periods of time which are extremely long compared to other human affairs. A solution lies in placing the relatively small volumes of the most hazardous of radioactive wastes deep within stable geologic formations. Some such formations are known to have been stable for hundreds of millions of years, and there is every reason to believe they will continue to be so for further geologic periods.

The ERDA Fiscal Year 1977 budget provides for a major expansion of the program for evaluating different kinds of geologic formations, and specific sites, in different parts of the continental U.S. The program will include involvement of knowledgable state government and university scientists and has a target of beginning test emplacement of a realistic inventory of commercial HLW in the mid-1980's, preferably on a regional basis. Only after successful completion of a test period would the waste be considered to be permanently in place. The present state of technology in the fields of earth sciences and nuclear facility engineering is considered adequate to reach this target. Geologic repositories acceptable for HLW disposal could also be used for TRU-contaminated waste, since requirements for the former would be more stringent.

Another segment of the program is to put the wastes into chemical and physical forms that would provide optimum safety during transport and storage. A substantial ERDA research and development effort is devoted to this, with the target of having the technology ready for a commercial-scale plant to be built and operating by 1983.

Still another program is directed toward the development of interim storage sites if federal custody of wastes is required before geologic isolation is available. A considerable effort to resolve this issue was begun in 1972. Conceptual designs have been completed on three engineering approaches (a water-filled basin, an air-cooled vault, and massive casks for each waste canister) and ERDA safety studies indicate that an HLW storage facility using any of these approaches could be operated at any of several large ERDA nuclear sites with minimal environmental impact. Construction of such a repository has been deferred and ERDA now believes that in view of delays in generation of commercial HLW, and the planned acceleration of the geologic site development program, the eventual need for surface storage will be much less than was thought when the program started. ERDA's own TRU-contaminated defense wastes have been stored retrievably for some years and the methods could be used for comparable commercial TRU waste if needed.

Low-level radioactive waste is generally considered to be adequately isolated by relatively shallow burial. Six commercial sites now provide burial of so-called low-level solid radioactive waste. ERDA does not perform R&D in support of these

sites but does have a program of studies on improved site selection and operation practices for equivalent ERDA burial grounds, and the results of these studies should be applicable in industry.

III. Defense and Commercial Waste Management Portion of FY 77 Presidental Budget

WASTE MANAGEMENT (ERDA)
(Dollars in Thousands)

		Actual FY 1975	Estimate FY 1976	Estimate Transition Quarter	Estimate FY 1978	
A.	Production Reactor Waste	$35,307	$43,250	$11,640	$51,260	
B.	Non-Production Reactor Waste	1,794	3,640	1,075	4,840	(1)
C.	Process Development	2,566	3,045	945	4,765	
D.	Supporting Services - Interim	1,470	930	460	645	
E.	ERDA Radioactive Waste R&D	8,933	13,720	3,980	21,050	
F.	Storage Operations & Related Activities	3,870	4,900	1,360	7,000	(2)
G.	Supporting Services - Long-Term	0	0	0	2,240	
	Total Waste Management (ERDA)	$53,940	$69,485	$19,460	$91,800	

(1) Interim Waste Management.

(2) Long-Term Management.

WASTE MANAGEMENT (COMMERCIAL)

A.	Terminal Storage R&D	$ 3,394	$ 4,580	$ 1,420	$33,700	$38,200
B.	Waste Processing R&D	4,097	5,355	1,450	19,870	20,470
C.	Supporting Studies and Evaluations	1,888	1,990	520	6,400	6,900
	Total Waste Management (Commercial)	$ 9,379	$11,925	$ 3,390	$59,970	$65,570

APPENDIX C

THE NRC WASTE MANAGEMENT PROGRAMS

Introduction

The Nuclear Regulatory Commission (NRC) has responsibility under the Atomic Energy Act for regulating management of nuclear wastes at licensed facilities, and disposal of these wastes. Under the Energy Reorganization Act, NRC was given additional licensing responsibility for long-term storage and disposal of high level radioactive wastes arising from both ERDA and commercial operations. In addition, the National Environmental Policy Act (NEPA) requires the NRC to assess environmental impacts associated with the regulation of waste and waste management activities at licensed facilities.

The NRC Waste Management Program

Recognizing that priorities needed to be reassigned to meet the growing need for objective waste management goals and effective programs to meet those goals, the Commission moved to develop an overall waste management program. Initial stages have been completed, tentative regulatory needs have been established, and a program is underway to meet these needs.

The Commission's nuclear waste management program is designed to address four major objectives:

- To provide objective performance goals (technical, social, economic, and environmental) against which nuclear waste management programs and strategies can be evaluated.

- To provide a framework of regulations, standards, and guides for management of nuclear wastes within which NRC can effectively and efficiently carry out the functions dictated by its responsibility to protect the public health and safety. This framework will have to be supported by a comprehensive series of environmental impact statements.

- To develop a methodology for implementing its goals and regulations and the data base needed to make effective use of this methodology.

- To be capable of performing licensing reviews on proposed waste management systems as required to meet its responsibilities.

Programs in Progress

Waste Management Goals

One of the highest priority items in the Commission's waste management program is the establishment of objective goals to point the direction for subsequent program efforts. To this end, a multidisciplinary Goals Task Force was established to address the many concerns requiring consideration in establishing waste management goals from the varied occupational and academic perspectives from which these concerns can be viewed. Concerns addressed by the task force include: magnitudes and lifetimes of potential hazards; technologies necessary for management of the wastes; institutions and institutional arrangements required for such management; interference with utilization of other resources; foreclosure of future options; impact on decisions and/or actions of future generations; time frames for action (e.g., immediate actions, horizons for prediction, perceived limits for commitment or prediction, etc.); distribution of hazards and benefits (geographical and temporal); uncertainties that will remain during decisionmaking; and public involvement in decisionmaking.

The Goals Task Force report is expected in the Fall of 1976.

Development of Waste Management Regulations

To provide effective guidance (planning base) for ERDA and the nuclear industry, new regulations will be structured to require conformance with a fixed set of minimum acceptable performance standards (technical, social, and environmental) for waste management activities while providing for flexibility in technological approach.

The regulations must be directed toward achieving the following goals:

- Isolation of radioactive wastes from man and his environment for sufficient periods (in some cases hundreds of thousands of years) to assure public health and safety and preservation of environmental values.

- Reduction to as low as reasonably achievable: 1) risk to the public health both from chronic exposure and possible accidental releases of radioactive materials from waste storage, processing, handling or disposal operations; and 2) long-term social commitments (land-use withdrawal, resource commitment, surveillance requirements, committed site proliferation, etc.)

Some key elements in the Commission's regulation development program for waste management presently under development are as follows:

- Development of new portions of Commission regulations to deal specifically with the licensing and regulation of waste management facilities and activities is underway. Also in preparation is a framework into which subsequent regulations concerned with waste management could be placed as they are adopted by the Commission. Preliminary staff program plans indicate that the framework should be published for public comment in the summer of 1977.

- Development of performance criteria for solidified HLW to ensure public safety during handling, transportation, storage, and disposal is in progress. Notice of the Commission's intent to publish performance criteria for HLW solid matrices, an overview of what these criteria might consist of, and solicitation for public comment on the criteria is planned for publication in the Federal Register in December 1976. The proposed rule, along with an assessment of its environmental impact, would follow in the summer of 1977.

- Criteria that will allow classification of wastes according to the degree of confinement necessary to ensure their decay to some acceptable low-risk level are being developed. A task force will recommend to the Commission waste classification criteria and definitions suited to the regulation of radioactive wastes. A proposed rule setting forth classification criteria is tentatively scheduled for publication in late 1977.

- A program to develop site selection criteria for HLW repositories will first define the earth-science parameters that must be considered in evaluating potential sites for deep geological HLW repositories. This preliminary definition is scheduled for completion in October 1976. The second-phase effort is directed toward developing acceptable earth-science criteria for use in determining site suitability, and developing other acceptance criteria including geographic, demographic, and socioeconomic factors. Preliminary staff estimates indicate that publication of the proposed site selection criteria, along with a supporting environmental impact statement, will occur in mid-1978.

- Efforts have been initiated to develop a criterion of acceptable risk for use in evaluating the effectiveness of proposed waste management, handling, transportation, storage, and disposal schemes.

Additional Studies Underway

A number of studies are underway to provide a data base and methodologies for longer-range actions having completion dates within two to five years.

- NRC is developing both a methodology and supporting data base needed to make independent assessments of systems proposed by industry or ERDA. One program will develop a methodology for assessing risk associated with long-term geologic isolation (disposal) of high-level and transuranic wastes.

- NRC is assisting the U.S. Geodetic Survey in its program on commercial burial sites by providing analytical services for samples obtained by USGS. This program is aimed at determining the processes and underlying principles controlling radioactive waste migration through soil. Results of this study will be used in the NRC's assessment of present waste burial practices.

- A study to characterize the types of solidified low-level waste generated in the nuclear fuel cycle is concerned with solidified evaporator concentrates and solid wastes generated as byproducts of liquid radwaste treatment systems in LWR's. Results from the study will be used as a technical basis for establishing criteria and standards for the storage, transportation, and disposal of low-level radioactive waste.

- NRC will prepare a Generic Environmental Impact Statement (GEIS) on the management of uranium mill tailings which will lead to the formulation of Commission regulations in this area. A draft is scheduled for publication in August 1978.

- Studies directed at developing comprehensive engineering information on the technical status, safety aspects, and costs associated with the decontamination and decommissioning of nuclear facilities are in progress for each major type of fuel cycle facility. Each study includes: (1) characteristics of the plant and site; (2) acceptable decontamination levels; (3) radiation exposures to workers and the general public; (4) a benefit/cost analysis; and (5) identification of research and development needs. Results from the study will be used to develop regulations and standards directed toward managing the wastes removed from or remaining within such facilities. The studies are scheduled for completion in 1978.

- In the last decade there has been a revolution in understanding of global geological processes. Beginning with a conceptual model of these processes, NRC will attempt to identify specific characteristics such as geologic stability and lack of usable resources which make particular sites suitable for disposal of nuclear wastes.

- The NRC has noted the range of questions regarding the desirability and possibility of further partitioning (i.e., separating) of TRU elements from the nuclear wastes. The proceedings of a workshop on this subject were made available to the public in August 1976.*

- The NRC has initiated an effort that will eventually lead to rulemaking proceedings on the disposition of approximately 600,000 gallons of liquid HLW now in storage at the NFS reprocessing plant site. In a Federal Register notice dated April 19, 1976, NRC announced the availability of a report entitled, "Alternative Processes for Managing Existing High Level Radioactive Wastes," which evaluates the possible application to NFS of waste management alternatives and processes under development for ERDA high-level wastes. A proposed regulation specifically governing the management of NFS wastes is required by 10 CFR 50, Appendix F.

*"The Management of Radioactive Waste: Waste Partitioning as an Alternative," Seattle, Wash., June 8-10, 1976, NR-CONF-001, U.S. Nuclear Regulatory Commission, Wash., D.C.

Planned Waste Management Programs

The need for additional studies is being assessed as program activities progress, a data base is compiled, and the regulatory framework takes shape. Some of the items currently in planning are as follows:

- A study on alternatives to the disposal of wastes by shallow land burial is being planned to guide the Commission in assessing the need for future regulations governing the disposal of low-level waste. The study is expected to yield results in approximately two years.

- A study to determine the design requirements for HLW repositories is planned to provide a basis for the development of standards and staff review methodologies.

- NRC staff is identifying requirements for model development, review methodology development, and specific site testing requirements to allow a timely and independent evaluation of HLW repositories and potential repository sites.

- A study is planned to assess the value of co-locating nuclear facilities and disposal sites to reduce the potential for transportation-related accidents.

- Preliminary plans have been made to study the technical aspects and the benefits/impacts of storing Kr-85 from fuel reprocessing facilities instead of releasing it to the atmosphere.

Summary

The safe management of nuclear wastes is one of the issues noted by many in the public, industry, and government as pivotal in the future role of nuclear power. The NRC has recognized this issue since its inception in January 1975. The priorities for NRC work in waste management are consistent with the needs of the overall national waste management program. Because present licensing procedures and criteria are adequate for the short term, priority attention is being given to the longer term, when the quantities of waste to be managed will be greater and licensing demands will increase.

Recognizing that its decisions will affect industry, other governmental jurisdictions, private interest groups, and the public at large, NRC has encouraged and will continue to encourage their participation in planning our program.

In summary, the NRC has firmly established waste management as a high-priority effort and has made the commitment to act rapidly and methodically to establish a sound regulatory base for licensing waste management activities. The technology for most waste management is in existence, and it is the implementation of that technology toward which the federal effort is now bent. The NRC regulatory approach is consistent with those circumstances.

APPENDIX D

EVOLUTION OF WASTE MANAGEMENT AND REPROCESSING POLICY AND PHILOSOPHY

The following analysis of waste management and reprocessing activities is done in the context of the evolution of both the technologies for and policies regarding those activities. While many past actions of industry and government agencies can be seen in retrospect as failures, many others are clearly successes. What we write today can only be judged on the basis of present policies and philosophy regarding these nuclear fuel-cycle activities, and in light of developments which have brought us to those policies.

In the development of reprocessing technology and policies, one major trend is apparent. This trend is driven by the availability of the Purex process which chemically separates spent fuels from reactors into three streams: unused uranium, plutonium (for fuel or weapons), and fission products (plus other unwanted materials which are wastes). The U and Pu have always been considered valuable materials.

Management of nuclear wastes has two separate but interactive trains of development. The first grew from the military program wherein the wastes from reprocessed fuel were residual materials requiring care, but whose care was considered peripheral to the urgent need for nuclear weapons and nuclear-powered ships. Competing priorities were intense and little attempt was made to optimize waste handling.

The philosophy regarding commercial nuclear wastes developed separately, though not independently, from policies regarding military wastes. Competing priorities here were fiscal, and the urgency not as severe.

The following pages sketch briefly the major steps in the development of policies regarding commercial waste management, the effect of military waste management on the commercial program, and decisions regarding reprocessing of commercially generated spent fuel.

WASTES FROM THE MILITARY PROGRAM

From the beginning of plutonium-weapon programs, it was recognized that the radioactive isotopes which were byproducts of those programs require careful stewardship and containment. Still, the foremost priority was to produce plutonium. While systems were designed and built to contain highly radioactive wastes, materials with less activity were consigned to shallow burial grounds using methods similar to those used for sanitary landfill, though with much greater care being exercised.*

*In the earliest days of the program, some low-level wastes were disposed of at sea. In the 1960's this practice was stopped by the United States.

Burial grounds for low-level military wastes have performed more or less as expected, and the same approach has been adopted for the disposal of commercially generated low-level wastes (LLW). Only in management of high-level wastes (HLW) from the military programs has there been a significant divergence between military and commercial policies.

ERDA operates three facilities--the Hanford Plant in Washington; the Savannah River Plant in South Carolina, and the National Reactor Testing Station (now INEL) in Idaho--to produce plutonium for the weapon program or to process irradiated fuel from experimental and naval reactors. As of 1974, all of these had produced some 85 million gallons of wastes in the forms of liquids, salt cake, sludges, crystals, and calcined granules.

At Hanford, early waste streams were neutralized and stored in single-walled carbon-steel tanks. (Double-walled tanks are now being placed in use.) Non-boiling wastes are converted to salt cake in the tanks. Self-boiling wastes are first fractionated to remove the long-half-life heat-generating isotopes of cesium and strontium. The residual waste is nonboiling and is being solidified.

At Savannah River, neutralized waste solutions are stored in carbon-steel tanks within carbon-steel shells. At Idaho, the wastes, initially stored in stainless-steel tanks, are calcined and then put in stainless-steel bins housed in concrete structures, from which they can be retrieved.

DEVELOPMENT OF POLICY FOR COMMERCIAL WASTES

In a climate of some urgency to develop a commercial reprocessing industry, a plan for the management of wastes at the West Valley plant of Nuclear Fuel Services, Inc., in New York, paralleled the technology used at Hanford and Savannah River, with some improvements in tank design and construction, and the recognition that something better would be required in the longer term.

The basic conceptual framework for management of civilian HLW emerged from a report by the National Academy of Science's Committee on Waste Management in 1957. The Committee noted that "the most promising method of disposal of high-level waste at the present time seems to be in salt deposits."[1]

On the basis of this approval, a research program was undertaken, a major portion of which, called Project Salt Vault, was to determine the consequences of exposing bulk salt to radiation and heat.[2] The site of the experiment was an inactive salt mine near Lyons, Kansas. Spent test-reactor fuel elements were used to represent solidified waste because the latter was not available at the time. Electric heaters were used in some experiments to supplement the thermal output of radioactive waste.

While this effort was underway, a fire at the AEC weapon facility at Rocky Flats, Colorado, gave rise to a large volume of low-level plutonium-contaminated debris. Such wastes had been routinely shipped to the Idaho Reactor Testing Station (now the

Idaho National Engineering Laboratory) for disposal in the AEC-operated burial ground there. Public officials in Idaho, however, were unwilling to accept the permanent stewardship of this new material, and also demanded the removal of all plutonium buried there in the past. The AEC agreed to begin the removal of waste from the Idaho burial grounds as soon as a repository became operational.

During the same period, steps were being taken by the AEC to formalize a regulatory policy concerning commercially generated wastes. Up to that time, whatever policy existed had been more or less ad hoc. The first systematic attempt to develop a waste management policy led ultimately to the adoption of Appendix F to 10 CFR 50. Among other provisions, the regulation provided that solidified wastes shall be "transferred to a federal repository no later than 10 years following the separation of fission products from the irradiated fuel." Thus, the Rocky Flats fire and the now officially acknowledged need for a repository caused the AEC to transform the Kansas salt mine from an experiment to a prototype HLW repository.[3]

Later, the safety of using the mine for such a purpose was questioned by a group of Kansas geologists and other scientists. Soon the issue also became political, with the result that Congress passed an amendment to the 1972 AEC Authorization Bill which prevented the AEC from implementing the Waste Repository Project until an advisory commission certified that the project was safe.[4] Meanwhile, a report by a commercial company was published revealing that hydraulic fracturing operations conducted in 1965 in a salt mine a few miles south of the Lyons project had resulted in an unexplained loss of water. This report raised questions about the possibility of voids in the rock and dissolution of the salt in and around the AEC's "repository," with consequent contamination of the ground in that area. This possibility was sufficiently credible that the repository project was dropped* and in February, 1972, it was officially cancelled.[5]

Several other ways of disposing of long-lived HLW were now considered, including such possibilities as space disposal and transmutation. Though some of these ways are still possibilities, such as disposal in the deep seabed, it was recognized that a new practical concept had to be developed for the short term. The notion of a Retrievable Surface Storage Facility (RSSF) was developed. It was envisioned that mausolea would be constructed in the West for the storage of wastes generated both by the AEC and commercial nuclear plants, pending transporation to a permanent disposal area.

This policy was announced by the AEC in May 1972, and in September 1974, a draft environmental statement on the RSSF was circulated. Resulting comments from environmental groups and from state and local governments were generally critical. The strongest voice was that of the Environmental Protection Agency, which said, among other things, "A major concern...is the possibility that economic factors could later dictate utilization of the facility as a permanent repository, contrary to the stated intent to make the RSSF interim in nature."[6] The request for funds to build an RSSF

*On the basis of this experience, site selection is now designed to avoid proximity to any drilling or mining.

was thereupon withdrawn by the AEC.[7] Although still a part of the ERDA R&D plan, the RSSF will be designed and partially tested, but built only if other repository plans fail (see Appendix B).

During 1975 and 1976, the federal agencies returned to the idea of geological disposal of high-level and transuranium-contaminated wastes as both their goal and their major programmatic thrust. With its approval of the 1977 budget for ERDA and NRC, Congress both endorsed and enhanced that approach. The goal now is to have two geologic repositories licensed and operating by 1985 when the first shipments of HLW are anticipated from commercial reprocessors. These programs are described briefly in Appendices B and C.

Two themes run through the history of the nation's development of a waste management policy. The first, still widely prevalent, is that technological expertise is alone sufficient to solve the waste management problem. The second is that the consideration of nontechnological problems is not only irrelevant, but in many cases is actually a hindrance to technological progress. There seems to have been an underlying belief that a waste management system would be self-implementing or automatically implemented when needed.

Because nuclear wastes appear primarily at the end of the nuclear fuel cycle, and because they appear commercially unprofitable, the area of wastes has not received great administrative attention. Authority, and therefore responsibility, has been diffused. The AEC created a Division of Waste and Scrap Management in 1970; a year later a stronger Division of Waste Management and Transportation was formed, but budget allocations remained small.

In late 1974, the AEC was dissolved and its functions distributed between the newly formed Energy Research and Development Administration, which took over the nuclear promotional functions of the AEC, and the Nuclear Regulatory Commission, which took up and expanded upon the duties of the AEC's Regulatory Division. The stage was now set for proper attention to be paid to what has become a major issue in the public controversy over whether nuclear power should be used for the generation of electricity.

The orientation of the Waste Management Program of the NRC is to deal with both technological and nontechnological problems attendant upon the safe and acceptable disposition of radioactive wastes. The attitude is that wastes can be safely disposed of, though they are recognizably dangerous if treated cavalierly. The public must ultimately decide on issues regarding the use of nuclear power, and adequate avenues must be offered to the public for learning both the problems and the benefits of an efficient waste management system. The administrative aspects of, and the implementation process for, waste management have now been recognized as important by all government agencies, and this attitude furnishes a strong focus for the NRC.[8]

At present all known potential disposal methods have been investigated, at least to the extent where they could be rejected for technical reasons. Salt formations and other geological formations are being investigated and evaluated throughout the U.S. for the disposal of commercially generated TRU-contaminated wastes, including HLW. Experimental work on the RSSF concept is being conducted at Hanford (Sec. 4.2.5). There is an ongoing international effort to determine whether and where in the ocean bottoms it may be practical to terminally dispose of high-level radioactive wastes.[9] The prospects of under-earth disposal in geologic media other than salt are being actively pursued at the Office of Waste Isolation at Oak Ridge (see App. B).

Low-level wastes have been buried in trenches on six commercial sites (Sec. 4.7). These operations are licensed by NRC or by states which have assumed that responsibility under agreements with NRC. Since the practice of burying radioactive materials in shallow trenches has been under question for some years, the AEC in 1974 proposed a rule forbidding the burial of materials containing significant quantities of TRU-contaminated wastes.

More recently, several events resulting in release of noticeable (but so far innocuous) levels of radioactive material outside reservation boundaries have led the federal government to reevaluate present practices. ERDA, NRC, and the USGS now have ongoing programs of review, research, and criteria development relative to the shallow land burial of LLW.[10] The NRC is considering the preparation of a generic environmental statement regarding that practice, in which alternatives to such burial will be addressed.

In May 1976 ERDA published "Alternatives for Managing Wastes from Reactors and Post-Fission Operations in the LWR Fuel Cycle" (ERDA-76-43). In this document are described many alternative technologies now available or nearly available for management of several kinds of nuclear wastes. While the technologies are not evaluated in terms of safety and environmental effects (this will follow in the ERDA GEIS on the subject), there is now at least a catalogue of technologies from which to choose. It is the implementation of the best available technologies toward which the federal programs (including those of the NRC) are directed today.

The recognition has been made explicit that present-day society--the benefitters from nuclear power--cannot commit its successors to the active management of wastes from the nuclear industry. We must take care of these wastes today with the best technology available in a way that will impose the fewest requirements for action on succeeding generations. Marcus Rowden, Chairman of the NRC, said in May 1976:

> "We cannot put off those decisions or actions to future generations or to unknown technologies. We who reap the current benefits of nuclear power, must assume and fulfill responsibility for the effective handling of the problem of nuclear wastes."

This is the philosophy that is now prevalent in the federal program, and that was used to select the system discussed in this Supplement.

References For Appendix D

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3. Solid Radioactive Wastes: Long-Term Storage in Central Kansas Salt Mine, AEC 180/87, June 12, 1970.

4. AEC Authorizing Legislation for Fiscal Year 1972, Project 72-3.

5. SECY-2271, "High Level Waste Management," February 2, 1972.

6. EPA response to WASH-1539 (Draft), November 15, 1974 (unpublished).

7. Letter to Hon. J. O. Pastore from R. L. Seamans, Jr., April 9, 1975.

8. Statement of Marcus A. Rowden, Chairman, USNRC, before the Joint Committee on Atomic Energy, on Nuclear Waste Management, May 12, 1976.

9. The Steering Committee for Nuclear Energy, "Progress Report on Work in the Field of Radioactive Waste Management," Organization for Economic Cooperation and Development, Nuclear Energy Agency, Paris, September 2, 1976, p. 8.

10. Statement of R. E. Cunningham, Acting Director, Division of Fuel Cycle and Material Safety, Office of Nuclear Material Safety and Safeguards, USNRC, before the Conservation, Energy, and National Resource Subcommittee on Government Operations, U.S. Congress, March 12, 1976.

11. Remarks by Marcus A. Rowden, Chairman, USNRC, before The International Symposium on the Management of Wastes from the LWR Fuel Cycle, Denver, Colo., July 12, 1976.

APPENDIX E

HISTORY OF TANK STORAGE

Underground storage tanks have been used for the storage of liquid HLW since operation of the first fuel reprocessing plants during World War II at Hanford. Since that time, designs of HLW storage tanks have gone through an evolutionary process. The development of designs at Hanford, the Savannah River Plant, and the Idaho Chemical Processing Plant is described briefly.

Hanford Tanks

Tanks constructed at Hanford[1] before 1952 were vertical concrete cylinders about 75 feet in diameter, with a flat bottom and a self-supporting domed roof covered with 6 to 9 feet of earth (Fig. E.1). The tanks were lined with carbon steel on the bottom, and on the sides up to 16 or 24 feet, depending on the size. The waste stored in the tanks was neutralized to a pH of 8 to 10 and had very low heat-evolution rates (less than 0.15 Btu/gal/hr). There was no provision for waste circulation within the tanks, and the tanks were equipped with air-cooled condensers venting to the atmosphere.

Beginning in 1952, tanks of the same diameter and basic design but of greater capacity were built for storage of waste with higher heat-generating capacity (Fig.E.2). These tanks had a capacity of 1,000,000 gallons each, were equipped with air-lift circulators, and were designed to contain wastes generating heat at rates from 1 to 10 Btu/gal/hr, which is a rate great enough to maintain the waste at boiling temperatures for 1 to 10 years. The tanks were equipped with water-cooled condensers, and had provisions for either returning condensate to the tank or sending it to a disposal site.

In 1958 the first of 18 confirmed cases of leaking tanks occurred at Hanford.[2] In each case, the remaining contents of the leaking tank were pumped to another tank. (An additional 14 tanks have been removed from service because their integrity was suspect.)

Because of this experience, tanks constructed at Hanford since 1968 have been of double-wall design (Fig. E.3). Double-wall tanks intended for HLW contain air-lift circulators to agitate the waste. Tanks for nonboiling liquids and salt cake do not have circulators. The tanks are fabricated of ASTM-A515 Grade 60 steel which is heat-treated after fabrication. This steel exhibited the lowest corrosion rate of many considered. An annulus between the primary and secondary tanks collects any leakage from the primary tank, and equipment is installed to detect and pump out liquid from the annular region. Equipment is also provided to pump liquids and slurries between tanks, tank forms, evaporators, and production plants.

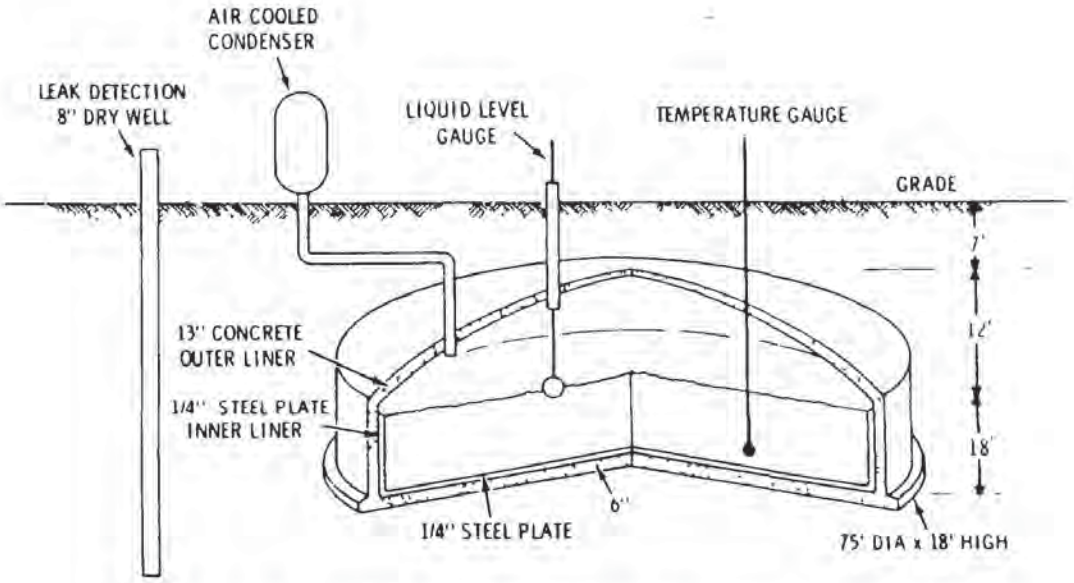

FIGURE E.1 - TYPICAL STORAGE TANK FOR NONBOILING WASTE

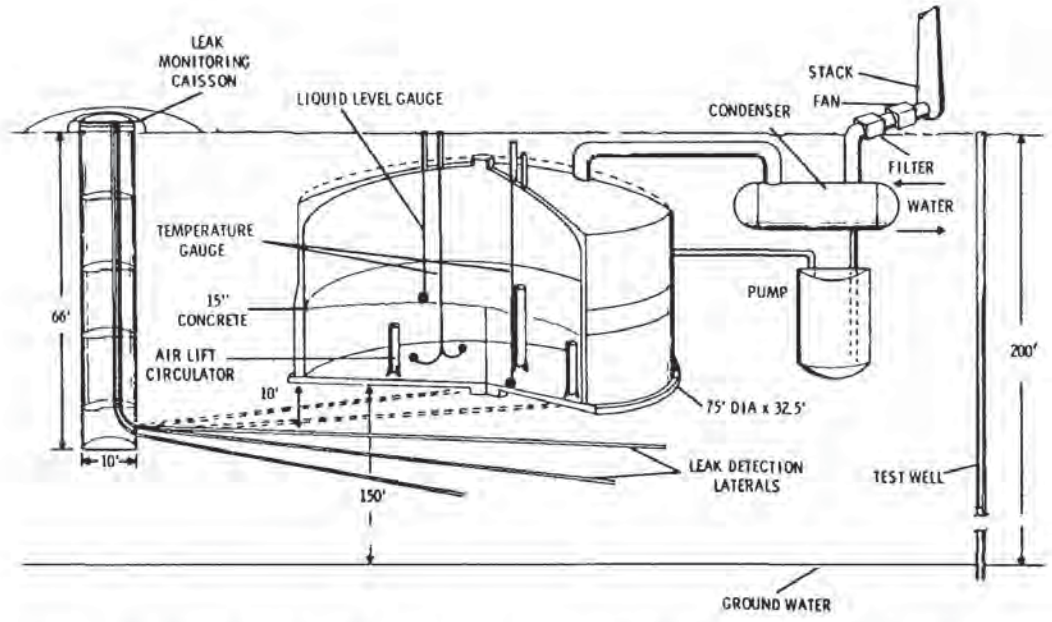

FIGURE E.2 - TYPICAL CONTROL AND SAFETY FEATURES FOR HIGH-LEVEL BOILING RADIOACTIVE WASTE STORAGE

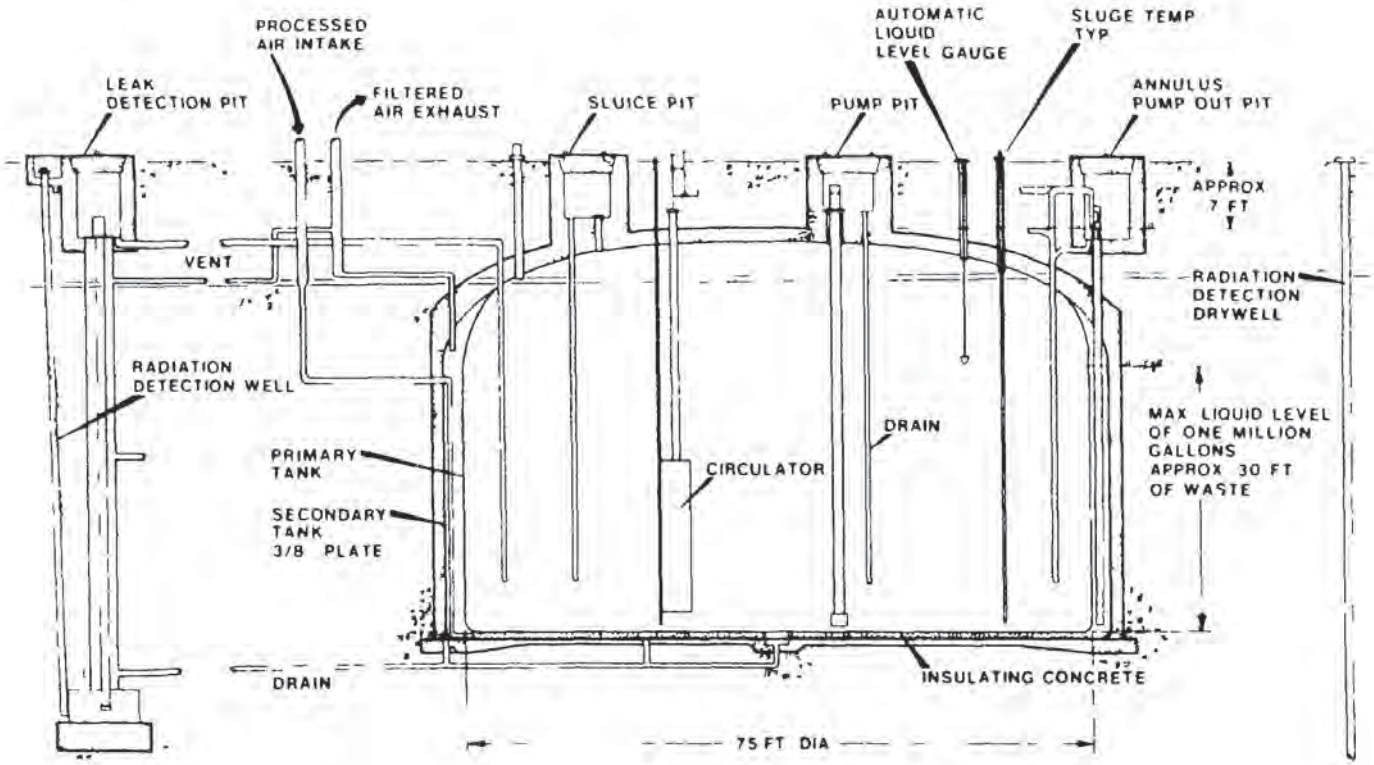

FIGURE E.3 - DOUBLE-WALL BOILING WASTE TANK

The 18 cases of leaking HLW tanks at Hanford have not produced any known effect offsite.[3] Leaks have varied from very small quantities up to 115,000 gallons from the 241-T-106 tank. A study carried out after the T-106 leakage incidents revealed that the deepest penetration of contamination was 89 feet below ground surface (115 ft. above the water table) and that about 880,000 ft^3 of sediments were contaminated.

Studies of a postulated "worst case" incident in which 800,000 gallons of waste were released from a storage tank[4] show that with conservative estimated travel times through the earth, radioactivity from the wastes will reach the Columbia River in about 22 years. Cesium and strontium never reach the river because of ground sorption. Other isotopes arrive at negligible levels; tritium and iodine because of the small quantities initially present; ruthenium and antimony because of radioactive decay, and antimony because of soil sorption.

Savannah River Plant Tanks

Waste tanks at SRP have been fabricated from carbon steel and built to three designs for self-heating waste and to another for low-heat waste that does not require cooling.[5]

Waste tanks designated as Type I[6] were the first constructed at SRP (in 1951-1953). They are vertical cylindrical tanks 75 feet in diameter by 24-1/2 feet high, of carbon steel, set in a carbon-steel pan 5 feet deep and 5 feet larger in diameter than the tank (Fig. E.4). The tank roof is supported by internal columns. The tank-and-pan assembly is in a reinforced concrete vault. The tank is equipped with cooling coils and equipment for determining whether there has been leakage into the pan annulus. All welds in both the pan and the primary tank were radiographed, defects were corrected, and repaired welds were rechecked.

Type II tanks[7] were constructed in 1955-1956. The primary container of these tanks consists of two steel cylinders, the inner 6 feet 8 inches in diameter, the outer 85 ft. in diameter, with a flat bottom and flat top. The inner cylinder houses a single supporting column for the vault roof. The primary tank again sits in a carbon-steel pan equipped with leak-detecting facilities. Cooling coils are provided in the tank. The primary tank was radiographically inspected, repaired when necessary, and rechecked, but the secondary pan was not radiographed.

The next group of tanks, built in 1958-1961, and designated Type IV,[8] were designed for storage of waste that does not require auxiliary cooling. The tanks are basically prestressed concrete cylinders lined with carbon steel, without cooling coils or secondary containment to catch leakage. Leaks are detected by a grid of channels in the concrete foundation, draining to a sump. All steel welds in the lining were radiographed.

Type III[9] tanks were the last constructed (1967-1972) (Fig. E.5). This design was developed after investigation of the leaks that had occurred in Type I and Type II tanks. The studies led to the conclusion that the primary leak-producing mechanism

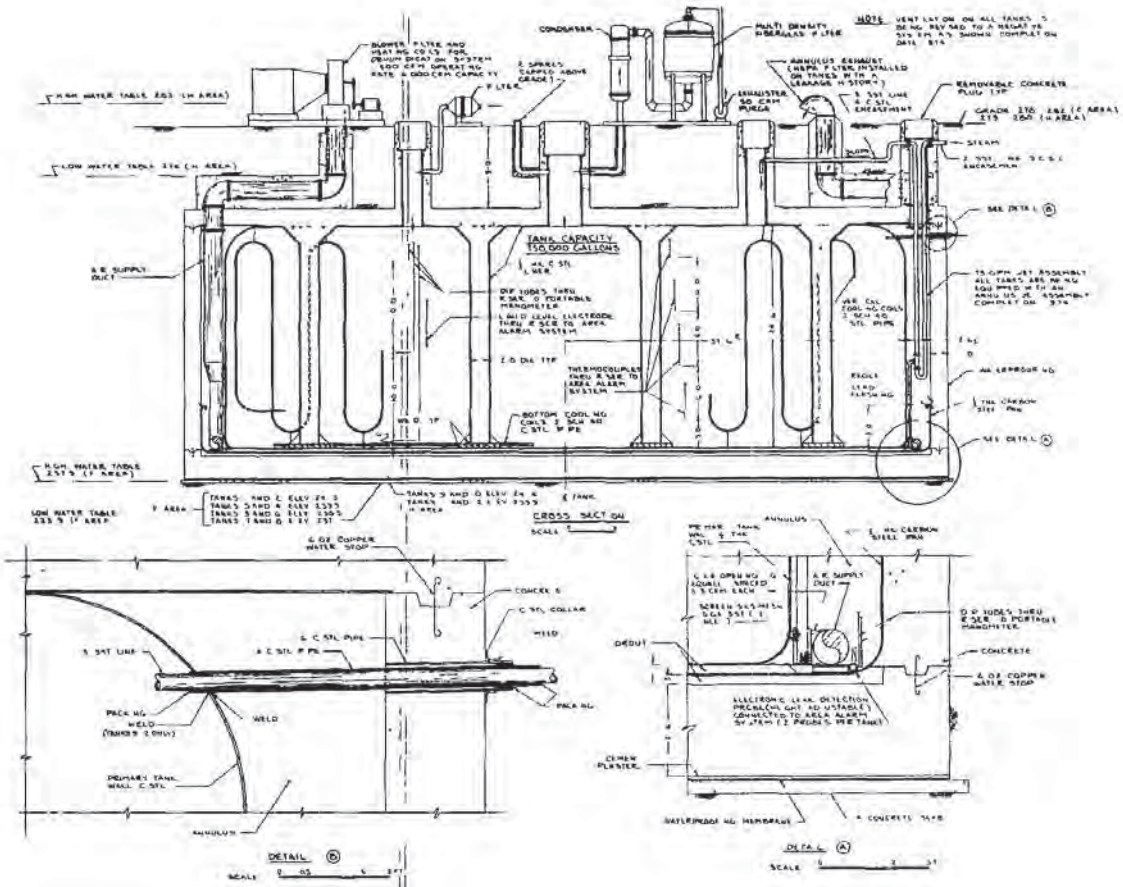

FIGURE E.4 - TYPE 1 WASTE STORAGE TANKS (1951-53)

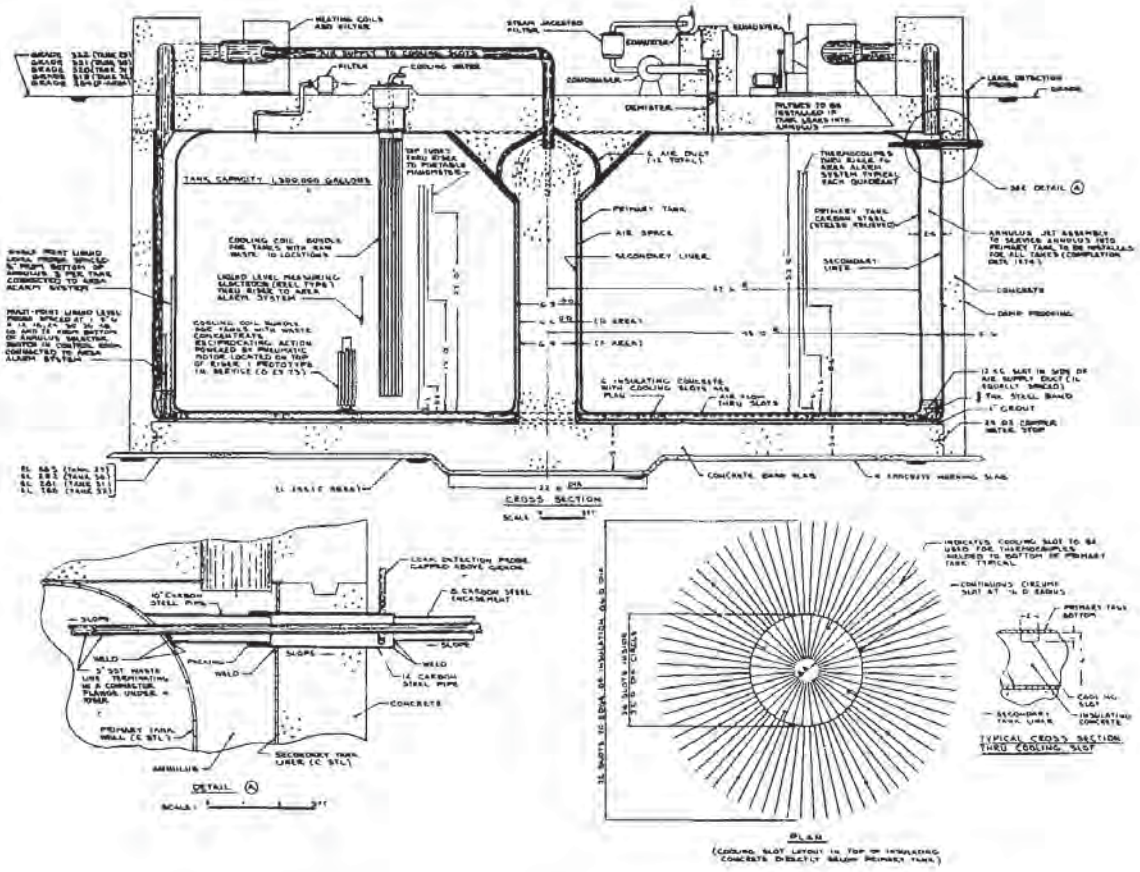

FIGURE E.5 - TYPE III STORAGE TANKS (1974-78)

was stress-corrosion cracking in or near weld seams, and that stress relieving after fabrication should eliminate cracking. Means were provided to relieve stresses generated during fabrication, and design changes were made in the roof-support column to minimize stresses in that area. These tanks also returned to the design concept of a tank set in a concrete vault, which in this case includes a carbon-steel liner to its full height. Original Type III tanks are equipped with removable cooling coils inserted through the roof. However, future tanks of this design will have permanent cooling coils similar to those in Type I and II tanks.

At SRP, 8 tanks have leaked radioactive waste from the primary tank into the annulus.[10] Four have leaked only a few gallons each and only one (Tank 16) has leaked waste into the ground. This was because the secondary pan filled with waste and overflowed before a transfer jet could be installed to empty the pan. Extensive study of that release indicates that no more than a few tens of gallons of waste reached the soil and that this waste migrated only a few feet from the vault.

Idaho Chemical Processing Plant Tanks

HLW generated by fuel recovery operations at ICPP are stored as an acid solution in 15 stainless-steel tanks.[11] Eleven of these have a volume of approximately 300,000 gallons each and 4 tanks which are no longer used are of 30,000-gallon capacity. The larger tanks are contained in concrete vaults, the tops of which are about 10 feet below ground. Eight of these tanks are equipped with internal cooling coils and reflux condensers.

The reinforced concrete vaults are equipped with sumps to contain any leakage from the tank. In addition, leakage from double-contained transfer lines or from diversion boxes drains to tank sumps. A few leaks have occurred in transfer lines.[12]

No leaks have occurred in any of the stainless-steel waste tanks at INEL since startup in 1951. Waste has entered two of the vaults as a result of accidental siphoning action following transfer of water from the sumps into the tanks. Water has also entered some vaults through the roof from heavy rain or melting snow. Surface drainage of the tank area is being improved in an effort to eliminate this in-leakage.

References For Appendix E

1. Final Environmental Statement, Waste Management Operations, Hanford Reservation, Richland, Washington, December 1975, Section II.1.1.2.2.2.
2. Ref. 1, Section III.2.2.
3. Ref. 1, Section III.2.2.2.
4. Ref. 1, Section III.2.2.3.
5. Draft Environmental Statement, Waste Management Operations, Savannah River Plant, ERDA-1537, July 1976.
6. Ref. 5, p. II-80.
7. Ref. 5, p. II-83.
8. Ref. 5, p. II-86.
9. Ref. 5, p. II-90.
10. Ref. 5, Appendix C.
11. Draft Environmental Statement, Waste Management Operations, Idaho Nuclear Engineering Laboratories, ERDA-1536, June 1976.
12. Ref. 11, Appendix 13.

APPENDIX F

THE TASK FORCE

As a result of the July 21, 1976 decision by the United States Court of Appeals for the District of Columbia Circuit (see Foreword), the Nuclear Regulatory Commission on August 13, 1976, issued a policy statement in which it directed establishment of a task force and the preparation of this Supplement. The events that followed were:

- 10-14 August: Experts in the areas of concern were contacted and commitments made for them to serve on the task force.

- 17 August: The task force assembled and assignments were made.

- 17 August - 8 September -- Drafts were prepared.

- 8-10 September: The task force met to review and discuss the drafts.

- 10-20 September: Second drafts were prepared and additional analyses were completed.

- 20-23 September: The task force met to produce the final working draft; a summary was written, and all sections were reviewed.

- 24-27 September: The final draft was assembled and edited.

- 28 September: The draft was sent to the Commission and to task-force members for final review.

Subsequent reviews and revisions led to publication in early October.

Participants

A large number of individuals took part in preparing this document, often at considerable personal inconvenience. Their contributions ranged from a massive literature search to the addition of technical ideas, to the careful perusal of each other's work, to the actual mechanics of publication.

Coordination and technical editing were the responsibilities of W.P. Bishop and F.J. Miraglia. Publication editor was P.L. Mead.

Technical contributions were made by more than 25 individuals who are listed below, together with the areas into which their written contributions fall. In alphabetical order, they are: E.L. Albenesius (TRU Interim Storage); J. W. Bartlett (Long-Term Risks); M. J. Bell (LLW); R. Bernero (Reprocessing); E. F. Branagan (Long-Term Risks); L. M. Bykoski (Socioeconomic Effects); F. M. Empson (Reprocessing and HLW Tank Storage); W. M. Hewitt (NRC Programs); V. Hodge (Sabotage); D. R. Hopkins (Transportation); M. S. Kearney (HLW Overview); F. D. King (Spent Fuel as a Waste); J. R. Kline (Biotic Effects); J. R. LaRiviere (HLW Interim Storage); R. C. Lincoln (Disposal); W. E. Lotz (ERDA Programs); P. E. McGrath (Long-Term Risks); A. S. Neuls (TRU-Combustible); S. J. Parry (The Fuel Cycle); R. Rhoads (Decommissioning); G. L. Richardson (TRU-Noncombustible); J. R. Roberts (Summary Tables); D. Rohrer (TRU-Combustible); W. A. Ross (HLW Solidification); R. E. L. Stanford (Decommissioning); K. G. Steyer (Decommissioning); and M. L. Wheeler (LLW Burial).

Review and comment were provided by all contributors, both during preparation of the document and after circulation of the drafts. In addition, there were special reviewers, who included J. O. Blomeke; M. C. Cullingford; L. C. Hulman; M. S. Kearney; W. E. Kreger; M. Kreiter; M. G. Malsch; I. C. Nelson; G. Sege; D. B. Shipler; H. F. Soule; and T. R. Workinger.

Document preparation was accomplished through the painstaking efforts of H. M. Gearin, M. S. Kearney, and P. L. Mead.

Support was provided on a daily basis and through many overtime hours by D. J. Corley; T. Botkin; R. D. DeMouy; A Haikalis; M. A. Jambor; A. E. King; R. Leslie; S. Lynd; K K. Pappas; C. S. Paul; and H. Stolzenberg.

The editors gratefully acknowledge the efforts of all the individuals named, and extend a special note of thanks to the staff of the Central Regulatory Electronics Stenographic System (CRESS) for the care with which they executed the seemingly endless revisions.

APPENDIX G -- SUPPLEMENTARY TABLES
(from GESMO and WASH-1248)

TABLE VIII (A)-5 From GESMO

ENVIRONMENTAL FACTORS FOR URANIUM-ONLY RECYCLE

ENVIRONMENTAL FACTORS	MINING	MILLING	CONVER-SION	ENRICH-MENT	UO2 FUEL FABRI-CATION	MOX FUEL FABRI-CATION	REACTOR	REPROCES-SING	TRANSPOR-TATION	WASTE MANAGE-MENT	SPENT FUEL STORAGE	TOTAL
RESOURCE USE												
ACRE-YRS OCCUPIED	2 1E+07	1 1E+06	1 2E+05	5 2E+04	1 4E+05	0 0	2 5E+06	1 3E+05	- -	8.0E+04	- -	2 5E+07
DISTURBED ACRES	2 8E+05	2 2E+04	1 3E+03	1 5E+03	2 2E+02	0 0	4 2E+04	1 2E+03	- -	1.6E+03	- -	3. 6E+05
COMMITTED ACRES	9 9E+03	2 7E+04	9 0E+01	- -	- -	0 0	8 3E+02	1 2E+03	- -	1.1E+01	- -	4.0E+04
WATER-GALLONS												
DISCHARGED TO AIR	- -	1 0E+12	1 2E+10	1 0E+12	- -	0 0	3 3E+13	2 2E+10	- -	5 4E+06	1 4E+10	3. 5E+13
DISCHARGED TO WATER	- -	- -	9 8E+10	5 6E+12	3 1E+10	0 0	3 3E+13	1 8E+11	- -	- -	- -	8. 9E+13
DISCHARGED TO GROUND	2 7E+12	- -	- -	- -	- -	0 0	- -	7 5E+08	- -	7 5E+07	- -	2 7E+12
TOTAL DISCHARGED	2 7E+12	1 0E+12	1 1E+11	5 7E+12	3 1E+10	0 0	6 6E+13	2 0E+11	- -	8.0E+07	1 4E+10	1 3E+14
BTU DISSIPATED	3 5E+14	1 3E+15	1 4E+14	1 1E+16	5 3E+12	0 0	2 7E+17	2 1E+14	5 4E+12	3 0E+13	1 1E+14	2 9E+17
TONS COAL	- -	- -	- -	3 7E+06	- -	0 0	- -	- -	- -	- -	- -	3. 7E+06
THERMS GAS	- -	1 1E+10	1 1E+09	- -	2 1E+08	0 0	- -	- -	- -	5.4E+07	- -	1. 2E+10
GALLONS FUEL OIL	1 8E+09	- -	- -	1 3E+07	- -	0 0	1 7E+10	7 5E+08	2 9E+07	4.0E+06	- -	2 0E+10
GWY ELECTRICITY	2 6E+00	6 0E+00	1 1E+00	1 4E+02	1 2E+00	0 0	2 4E+02	1 5E+00	- -	5.8E-02	4 2E-02	3 8E+02
COAL EQUIVALENT OF 2.2 GWY ELECTRICITY USED												
COAL BURNED (TONS)	6 0E+06	1 4E+07	2 7E+06	2 2E+08	2 7E+06	0 0	5 5E+08	2 5E+06	- -	1.3E+05	9 8E+04	9. 0E+08
SLUDGE (TONS)	8.3E+05	1 9E+06	3 7E+05	4 4E+07	3 7E+05	0 0	7 6E+07	4 9E+05	- -	1 9E+04	1 4E+04	1. 2E+08
SOX TO ATMOS (MT)	6 8E+04	1 6E+05	3 0E+04	3 6E+06	3 1E+04	0 0	6 2E+06	4 0E+04	- -	1 5E+03	1 1E+03	1. 0E+07
NOX TO ATMOS (MT)	5 5E+04	1 3E+05	2 4E+04	2 9E+06	2 4E+04	0 0	5 0E+06	3 2E+04	- -	1.2E+03	8 9E+02	8 2E+06
CO TO ATMOS (MT)	1 1E+03	2 5E+02	4 9E+02	5 8E+04	4 9E+02	0 0	1 0E+05	6 4E+02	- -	2.5E+01	1 8E+01	1 6E+05
PART TO ATMOS (MT)	3 2E+02	7 6E+03	1 5E+02	1 7E+05	1 5E+02	0 0	3 0E+05	1 9E+03	- -	7 4E+01	5 3E+01	4. 9E+05
HYDROCARBONS (MT)	5 5E+02	1 3E+02	2 4E+02	2 9E+04	2 4E+02	0 0	5 0E+04	2 2E+02	- -	1 2E+01	8 9E+00	8 2E+04
PLANT EFFLUENTS TO ATMOSPHERE (METRIC TONS)												
SOX	7 4E+04	4 9E+02	3 2E+04	4 6E+04	- -	0 0	4 8E+05	1 8E+04	7 9E+02	3 5E+01	- -	6 5E+05
NOX	6 1E+04	9 8E+04	1 5E+04	3 7E+04	- -	0 0	2 8E+05	7 2E+04	6 3E+02	4.1E+01	- -	5 7E+05
CO	- -	- -	- -	7 4E+02	- -	0 0	1 8E+04	1 4E+02	4 0E+02	2 5E+01	- -	2 5E+04
PARTICULATES	8 9E+03	4 9E+01	4 8E+02	2 2E+01	2 5E+04	0 0	6 4E+04	1 9E+01	1 4E+02	1.0E+01	- -	1 2E+05
NO2	- -	- -	4 4E+02	- -	- -	0 0	- -	- -	- -	- -	- -	4 4E+02
FLUORIDES	- -	- -	2 5E+02	1 4E+02	3 1E+01	0 0	- -	1 6E+02	- -	- -	- -	5 7E+02
HYDROCARBONS	4 4E+03	1 9E+04	2 1E+03	2 7E+02	- -	0 0	2 0E+04	1 9E+01	8 6E+02	- -	- -	4 9E+04
ALDEHYDE	- -	- -	- -	- -	- -	0 0	- -	- -	1 0E+01	4.0E-01	- -	7 0E+01
ORGANIC ACID	- -	- -	- -	- -	- -	0 0	- -	- -	8 6E+01	- -	- -	8 6E+01

TABLE VIII (A)-5 From GESMO (Continued)

ENVIRONMENTAL FACTORS FOR URANIUM-ONLY RECYCLE

ENVIRONMENTAL FACTORS	MINING	MILLING	CONVERSION	ENRICHMENT	UO2 FUEL FABRICATION	MOX FUEL FABRICATION	REACTOR	REPROCESSING	TRANSPORTATION	WASTE MANAGEMENT	SPENT FUEL STORAGE	TOTAL
PLANT EFFLUENTS TO ATMOSPHERE (CURIES)												
RN-222	2.1E+07	4.0E+06	3.2E-02	- -	- -	0 0	- -	- -	- -	2.4E+01	- -	2.5E+07
RA-226	- -	1.2E+01	9.4E-03	- -	- -	0 0	- -	- -	- -	- -	- -	1.2E+01
URANIUM	- -	4.7E+02	6.4E+00	3.3E+00	1.2E+00	0 0	- -	2.7E-01	- -	97.E-09	- -	4.8E+02
PU (ALPHA)	- -	- -	- -	2.8E-09	- -	0 0	- -	2.7E+00	- -	2.8E-01	- -	3.0E+00
PU-241 (BETA)	- -	- -	- -	- -	- -	0 0	- -	3.4E+01	- -	3.7E-01	- -	3.4E+01
TRANS-PU NUCLIDES	- -	- -	- -	- -	- -	0 0	- -	5.3E+00	- -	1.1E-03	- -	5.3E+00
H-3	- -	- -	- -	- -	- -	0 0	2.8E+06	5.9E+07	- -	2.4E-02	- -	6.2E+07
C-14	- -	- -	- -	- -	- -	0 0	4.2E+04	1.8E+04	- -	- -	- -	1.2E+05
KR-85	- -	- -	- -	- -	- -	0 0	2.1E+06	1.2E+09	- -	3.8E-01	2.8E+05	1.2E+09
SR-90	- -	- -	- -	- -	- -	0 0	- -	1.8E+01	- -	3.1E-02	- -	1.8E+01
TC-99	- -	- -	- -	1.8E+01	- -	0 0	- -	2.4E-02	- -	- -	- -	3.8E+01
I-129	- -	- -	- -	- -	- -	0 0	- -	1.1E+02	- -	- -	- -	1.1E+02
I-131	- -	- -	- -	- -	- -	0 0	6.0E+02	2.7E+02	- -	- -	- -	9.7E+02
OTHER RADIOACTIVITY	- -	- -	- -	6.2E-01	- -	0 0	5.4E+07	1.1E+07	- -	1.1E+00	- -	5.4E+07
PLANT EFFLUENTS TO WATER BODIES (METRIC TONS)												
SO4=	- -	- -	5.1E+04	3.3E+02	- -	0 0	1.4E+07	5.8E+01	- -	- -	- -	1.4E+07
NO3-	- -	- -	8.1E+02	3.7E+02	6.1E+02	0 0	- -	- -	- -	- -	- -	1.1E+04
CL-	- -	- -	1.4E+04	1.5E+02	- -	0 0	1.2E+05	2.9E+02	- -	- -	- -	1.2E+06
FLUORIDES	- -	- -	8.5E+02	3.7E+01	1.8E+02	0 0	- -	- -	- -	- -	- -	1.1E+02
NA+	- -	- -	4.7E+04	- -	- -	0 0	- -	5.8E+01	- -	- -	- -	4.7E+04
CA++	- -	- -	3.2E+03	- -	8.8E-01	0 0	- -	- -	- -	- -	- -	3.2E+03
NH3	- -	- -	1.6E+04	- -	6.9E+01	0 0	- -	- -	- -	- -	- -	1.6E+04
FE	- -	- -	2.1E+02	- -	- -	0 0	- -	- -	- -	- -	- -	2.1E+02
PLANT EFFLUENTS TO WATER BODIES (CURIES)												
TRANS-PU NUCLIDES	- -	- -	- -	- -	- -	0 0	- -	- -	- -	- -	- -	- -
PU (ALPHA)	- -	- -	- -	5.5E-07	- -	0 0	- -	- -	- -	- -	- -	5.5E-07
URANIUM	- -	- -	2.4E+02	2.3E-01	2.8E+02	0 0	- -	- -	- -	- -	- -	5.2E+02
TH-230	- -	- -	7.6E+01	- -	- -	0 0	- -	- -	- -	- -	- -	7.6E+01
RA-226	- -	- -	1.2E+00	- -	- -	0 0	- -	- -	- -	- -	- -	1.2E+00
I-129	- -	- -	- -	- -	- -	0 0	- -	- -	- -	- -	- -	- -
TC-99	- -	- -	- -	4.3E+02	- -	0 0	- -	- -	- -	- -	- -	4.3E+02
SR-90	- -	- -	- -	- -	- -	0 0	- -	- -	- -	- -	- -	- -
C-14	- -	- -	- -	- -	- -	0 0	- -	- -	- -	- -	- -	- -
H-3	- -	- -	- -	- -	- -	0 0	8.8E+05	- -	- -	- -	- -	8.3E+05
OTHER RADIOACTIVITY	- -	- -	- -	8.1E+00	- -	0 0	1.1E+03	- -	- -	- -	- -	1.2E+02

TABLE VIII (A)-5 From GESMO (Continued)

ENVIRONMENTAL FACTORS FOR URANIUM-ONLY RECYCLE

ENVIRONMENTAL FACTORS	MINING	MILLING	CONVER-SION	ENRICH-MENT	UO2 FUEL FABRI-CATION	MOX FUEL FABRI-CATION	REACTOR	REPROCES-SING	TRANSPOR-TATION	WASTE MANAGE-MENT	SPENT FUEL STORAGE	TOTAL
PLANT WASTE GENERATED (CUBIC METERS)												
CHEMICAL COMPOUNDS	- -	- -	1 6E+05	7 1E+02	1 5E+05	0 0	- -	2 3E+04	- -	- -	- -	3 3E+05
MILL TAILINGS	- -	6 9E+08	- -	- -	- -	0 0	- -	- -	- -	- -	- -	6 9E+08
TRANS-U SOLIDS	- -	- -	- -	- -	- -	0 0	- -	1 3E+05	- -	- -	- -	1 3E+05
HIGHLEVEL SOLIDS	- -	- -	- -	- -	- -	0 0	- -	6 5E+03	- -	- -	- -	6 5E+03
OTHER RAD SOLIDS	- -	- -	7 3E+04	6.5E+04	- -	0 0	3 8E+06	- -		- -	1 7E+03	3 9E+06
PERSON-REM COMMITMENT - OCCUPATIONAL												
TOTAL BODY	1 1E+06	5 0E+05	4 0E+03	3 5E+03	5 1E+04	0 0	2 3E+06	7 2E+04	5 4E+03	3 0E+03	5 6E+03	4 0E+06
G I TRACT	1 1E+06	1 9E+05	2 4E+03	1 8E+03	5 0E+04	0 0	2 3E+06	7 2E+04	5 4E+03	3 1E+03	5 6E+03	3 7E+06
BONE	1 6E+06	2 0E+06	4 7E+04	3 2E+04	6 6E+04	0 0	2 7E+06	7 2E+04	5 4E+03	3 3E+04	5 6E+03	6 1E+06
LIVER	1 1E+06	1 9E+05	4 9E+03	3 5E+03	4 9E+04	0 0	2 3E+06	7 2E+04	5 4E+03	5 0E+03	5 6E+03	3 7E+06
KIDNEY	1 6E+06	2 3E+05	1 2E+04	8 5E+03	5 4E+04	0 0	2 3E+06	7 2E+04	5 4E+03	4 4E+03	5 6E+03	4 2E+06
THYROID	1 1E+06	1 9E+05	4 0E+03	3 5E+03	5 0E+04	0 0	2 3E+06	7 2E+04	5 4E+03	3 0E+03	5 6E+03	3 7E+06
LUNG	5 8E+06	4 3E+06	3 1E+04	7 5E+04	2 0E+06	0 0	2 3E+06	7 2E+04	5 4E+03	2 5E+03	5 6E+03	1 5E+07
SKIN	1 1E+06	1 9E+05	1 1E+04	1 1E+04	5 0E+04	0 0	2 3E+06	7 2E+04	5 4E+03	2 5E+03	5 6E+03	3 7E+06
PERSON-REM COMMITMENT - OFF-SITE U S POPULATION												
TOTAL BODY	2 7E+06	5 2E+05	3 9E+04	1 3E+02	2 6E+03	0 0	3 1E+05	1 1E+06	1 6E+03	3 2E+00	1 4E+01	4 6E+06
G I TRACT	1 0E+05	2 0E+04	5 0E+03	7 3E+03	2 8E+03	0 0	3 0E+05	1 6E+06	1 6E+03	7.3E−01	1 4E+01	2 0E+06
BONE	8 7E+06	1 7E+06	9 2E+04	9 4E+02	4 2E+04	0 0	1 1E+06	2 6E+06	1 6E+03	1 6E+01	1 4E+01	1 4E+07
LIVER	2 2E+06	4 0E+05	6 2E+02	2 2E+02	5 2E+00	0 0	3 1E+05	1 1E+06	1 6E+03	3.2E+00	1 4E+01	4 0E+06
KIDNEY	1 0E+07	1 9E+06	1 1E+04	3 0E+03	6 8E+03	0 0	3 0E+05	1 1E+06	1 6E+03	1.1E+01	1 4E+01	1 3E+07
THYROID	6 5E+03	1 4E+03	4 3E+01	1 7E+01	4 7E+00	0 0	4 8E+05	1 9E+06	1 6E+03	3 8E−01	1 4E+01	2 4E+06
LUNG	8 1E+05	1 6E+05	8 8E+02	4 4E+02	1 5E+02	0 0	3 0E+05	1 2E+06	1 6E+03	1 4E+00	1 1E+01	2 4E+06
SKIN	6 5E+03	1 4E+03	3 4E+01	1 2E+02	4 7E+00	0 0	3 1E+05	6 6E+06	1 6E+03	3 8E−01	1 2E+03	6 9E+06
PERSON-REM COMMITMENT - TO FOREIGN POPULATION FROM U S INDUSTRY												
TOTAL BODY	- -	- -	- -	- -	- -	0 0	2 1E+05	7 0E+05	- -	- -	6 4E+01	3 1E+05
G I TRACT	- -	- -	- -	- -	- -	0 0	2 1E+05	7 0E+05	- -	- -	6 4E+01	3 1E+05
BONE	- -	- -	- -	- -	- -	0 0	1 0E+06	2 3E+06	- -	- -	6 4E+01	1 3E+06
LIVER	- -	- -	- -	- -	- -	0 0	2 1E+05	7 0E+05	- -	- -	6 4E+01	3 1E+05
KIDNEY	- -	- -	- -	- -	- -	0 0	2 1E+05	7 0E+05	- -	- -	6 4E+01	3 1E+05
THYROID	- -	- -	- -	- -	- -	0 0	2 1E+05	7 0E+05	- -	- -	5 4E+01	3 1E+05
LUNG	- -	- -	- -	- -	- -	0 0	2 1E+05	1 1E+06	- -	- -	1 4E+02	1 3E+06
SKIN	- -	- -	- -	- -	- -	0 0	2 5E+05	2 6E+07	- -	- -	5 4E+03	2 6E+07

TABLE VIII (A)-6 From GESMO

ENVIRONMENTAL FACTORS FOR NO RECYCLE

ENVIRONMENTAL FACTORS	MINING	MILLING	CONVERSION	ENRICHMENT	UO2 FUEL FABRICATION	MOX FUEL FABRICATION	REACTOR	REPROCESSING	TRANSPORTATION	WASTE MANAGEMENT	SPENT FUEL STORAGE	TOTAL
RESOURCE USE												
ACRE-YRS OCCUPIED	2.3E+07	1.2E+06	1.3E+05	5.2E+04	1.4E+05	0 0	2.5E+06	0 0	- -	7.1E+04	- -	2.8E+07
DISTURBED ACRES	3.2E+05	3.7E+04	1.3E+03	1.5E+03	2.3E+02	0 0	4.2E+04	0 0	- -	1.6E+03	- -	4.1E+05
COMMITTED ACRES	1.1E+04	3.1E+04	9.0E+01	- -	- -	0 0	8.3E+02	0 0	- -	1.1E+03	- -	4.5E+04
WATER (GALLONS)												
DISCHARGED TO AIR	- -	1.2E+12	1.3E+10	1.0E+12	- -	0 0	3.3E+12	0 0	- -	1.7E+07	1.1E+10	5.5E+13
DISCHARGED TO WATER	- -	- -	1.1E+11	5.5E+12	3.1E+10	0 0	3.3E+12	0 0	- -	- -	- -	8.8E+12
DISCHARGED TO GROUND	3.0E+12	- -	- -	- -	- -	0 0	- -	0 0	- -	6.1E+07	- -	3.0E+12
TOTAL DISCHARGED	3.0E+12	1.2E+12	1.3E+11	5.6E+12	3.1E+10	0 0	6.6E+12	0 0	- -	7.8E+07	1.1E+10	1.3E+14
BTU DISSIPATED	3.9E+14	1.4E+15	1.6E+14	1.1E+16	5.3E+12	0 0	2.7E+17	0 0	4.9E+12	3.5E+13	9.5E+12	2.9E+17
TONS COAL	- -	- -	- -	3.7E+06	- -	0 0	- -	0 0	- -	- -	- -	3.7E+06
THERMS GAS	- -	1.2E+10	1.2E+09	- -	2.1E+08	0 0	- -	0 0	- -	4.5E+07	- -	1.3E+10
GALLONS FUEL OIL	2.0E+09	- -	- -	1.3E+07	- -	0 0	1.7E+10	0 0	3.5E+07	3.2E+06	- -	1.9E+10
GWY ELECTRICITY	2.9E+00	6.7E+00	1.3E+00	1.4E+02	1.2E+00	0 0	2.4E+02	0 0	- -	4.4E-02	8.4E-02	3.8E+02
COAL EQUIVALENT OF 2.2 GWY ELECTRICITY USED												
COAL BURNED (TONS)	6.7E+06	1.6E+07	3.0E+06	3.2E+08	2.7E+06	0 0	5.5E+08	0 0	- -	1.0E+05	2.0E+05	9.0E+08
SLUDGE (TONS)	9.3E+05	2.2E+06	4.2E+05	4.4E+07	3.7E+05	0 0	7.6E+07	0 0	- -	1.5E+04	2.7E+04	1.2E+08
SOX TO ATMOS (MT)	7.6E+04	1.8E+05	3.5E+04	3.6E+06	3.1E+04	0 0	6.2E+06	0 0	- -	1.2E+03	2.2E+03	1.0E+07
NOX TO ATMOS (MT)	6.1E+04	1.4E+05	2.8E+04	2.9E+06	2.4E+04	0 0	5.0E+06	0 0	- -	9.3E+02	1.8E+03	8.1E+06
CO TO ATMOS (MT)	1.2E+03	2.8E+03	5.5E+02	5.8E+04	4.9E+02	0 0	1.0E+05	0.0	- -	1.8E+01	3.6E+01	1.6E+05
PART TO ATMOS (MT)	3.7E+03	8.5E+02	1.7E+03	1.7E+05	1.5E+03	0 0	3.0E+05	0 0	- -	5.6E+01	1.1E+02	4.9E+05
HYDROCARBONS (MT)	6.1E+02	1.4E+03	2.8E+02	2.9E+04	2.4E+02	0 0	5.0E+04	0 0	- -	9.3E+00	1.8E+01	8.1E+04
PLANT EFFLUENTS TO ATMOSPHERE (METRIC TONS)												
SOX	8.3E+04	5.5E+02	3.4E+04	4.6E+04	- -	0 0	4.8E+05	0 0	8.8E+02	2.9E+01	- -	6.5E+05
NOX	6.8E+04	1.1E+05	1.8E+04	3.7E+04	- -	0 0	2.8E+05	0 0	6.4E+03	3.3E+01	- -	5.2E+05
CO	-	- -	- -	7.4E+02	- -	0 0	1.6E+04	0 0	4.1E+03	2.0E+01	- -	2.3E+04
PARTICULATES	9.9E+03	5.5E+03	5.5E+02	2.3E+03	2.5E+04	0 0	8.4E+04	0 0	2.8E+02	8.0E-02	- -	1.2E+05
NH3	- -	- -	4.7E+02	- -	- -	0 0	- -	0 0	- -	- -	- -	4.7E+02
FLUORIDES	- -	- -	1.8E+02	1.4E+03	2.1E+01	0 0	- -	0 0	- -	- -	- -	4.5E+02
HYDROCARBONS	4.9E+03	2.2E+04	2.5E+02	2.7E+02	- -	0 0	2.0E+04	0 0	3.3E+02	- -	- -	5.1E+04
ALDEHYDE	- -	- -	- -	- -	- -	0 0	- -	0 0	5.5E+01	3.3E-01	- -	5.5E+01
ORGANIC ACID	- -	- -	- -	- -	- -	0 0	- -	0 0	9.6E+01	- -	- -	9.6E+01

TABLE VIII (A)-6 From GESMO (Continued)

ENVIRONMENTAL FACTORS FOR NO RECYCLE

ENVIRONMENTAL FACTORS	MINING	MILLING	CONVER-SION	ENRICH-MENT	UO2 FUEL FABRI-CATION	MOX FUEL FABRI-CATION	REACTOR	REPROCES-SING	TRANSPOR-TATION	WASTE MANAGE-MENT	SPENT FUEL STORAGE	TOTAL
PLANT EFFLUENTS TO ATMOSPHERE (CURIES)												
RN-222	2 4E+07	4 4E+06	2 4E-02	- -	- -	0 0	- -	0 0	- -	2 0E+01	- -	2 8E+07
RA-226	- -	1 3E+01	1 0E-02	- -	- -	0 0	- -	0 0	- -	- -	- -	1 3E+01
URANIUM	- -	5 3E+02	7 1E+00	3 9E+00	1 2E+00	0 0	- -	0 0	- -	7 9E-09	- -	5 4E+02
PU (ALPHA)	- -	- -	- -	- -	- -	0 0	- -	0 0	- -	2 3E-03	- -	2 3E-03
PU-241 (BETA)	- -	- -	- -	- -	- -	0 0	- -	0 0	- -	3 0E-02	- -	3 0E-02
TRANS-PU NUCLIDES	- -	- -	- -	- -	- -	0 0	- -	0 0	- -	9 0E-04	- -	9.0E-04
H-3	- -	- -	- -	- -	- -	0 0	3 8E+06	0 0	- -	- -	- -	3 8E+06
C-14	- -	- -	- -	- -	- -	0 0	4 2E+04	0 0	- -	- -	- -	4 3E+04
KR-85	- -	- -	- -	- -	- -	0 0	2 1E+06	0 0	- -	- -	5 6E+05	2 6E+06
SR-90	- -	- -	- -	- -	- -	0 0	- -	0 0	- -	2 5E-02	- -	2 5E-02
TC-99	- -	- -	- -	- -	- -	0 0	- -	0 0	- -	- -	- -	- -
I-129	- -	- -	- -	- -	- -	0 0	- -	0 0	- -	- -	- -	- -
I-131	- -	- -	- -	- -	- -	0 0	6 0E+02	0 0	- -	- -	- -	6 0E+02
OTHER RADIOACTIVITY	- -	- -	- -	- -	- -	0 0	5 4E+07	0 0	- -	9 0E-01	- -	5 4E+07
PLANT EFFLUENTS TO WATER BODIES (METRIC TONS)												
SO4=	- -	- -	5 4E+04	3 2E+03	- -	0 0	1 4E+07	0 0	- -	- -	- -	1 4E+07
NO3-	- -	- -	9 6E+02	3 6E+02	6 1E+02	0 0	- -	0 0	- -	- -	- -	1 1E+04
CL-	- -	- -	1 6E+04	1 5E+03	- -	0 0	1 2E+06	0 0	- -	- -	- -	1 2E+06
FLUORIDES	- -	- -	9 5E+02	3 7E+01	1 8E+02	0 0	- -	0 0	- -	- -	- -	1 2E+03
NA+	- -	- -	5 1E+04	- -	- -	0 0	- -	0 0	- -	- -	- -	5 1E+04
CA++	- -	- -	2 9E+03	- -	8 8E-01	0 0	- -	0 0	- -	- -	- -	3 9E+03
NH3	- -	- -	1 7E+04	- -	6 9E+01	0 0	- -	0 0	- -	- -	- -	1 7E+04
FE	- -	- -	2 2E+02	- -	- -	0 0	- -	0 0	- -	- -	- -	2 2E+02
PLANT EFFLUENTS TO WATER BODIES (CURIES)												
TRANS-PU NUCLIDES	- -	- -	- -	- -	- -	0 0	- -	0 0	- -	- -	- -	- -
PU (ALPHA)	- -	- -	- -	- -	- -	0 0	- -	0 0	- -	- -	- -	- -
URANIUM	- -	- -	2 8E+02	2 2E-01	2 2E+02	0 0	- -	0 0	- -	- -	- -	5 5E+02
TH-230	- -	- -	4 2E+01	- -	- -	0 0	- -	0 0	- -	- -	- -	4 2E+01
RA-226	- -	- -	1 4E+00	- -	- -	0 0	- -	0 0	- -	- -	- -	1 4E+00
I-129	- -	- -	- -	- -	- -	0 0	- -	0 0	- -	- -	- -	- -
TC-99	- -	- -	- -	- -	- -	0 0	- -	0 0	- -	- -	- -	- -
SR-90	- -	- -	- -	- -	- -	0 0	- -	0 0	- -	- -	- -	- -
C-14	- -	- -	- -	- -	- -	0 0	- -	0 0	- -	- -	- -	- -
H-3	- -	- -	- -	- -	- -	0 0	8 8E+05	0 0	- -	- -	- -	8 8E+05
OTHER RADIOACTIVITY	- -	- -	- -	- -	- -	0 0	1 1E+02	0 0	- -	- -	- -	1 1E+02

TABLE VIII (A)-6 From GESMO (Continued)

ENVIRONMENTAL FACTORS FOR NO RECYCLE

ENVIRONMENTAL FACTORS	MINING	MILLING	CONVER-SION	ENRICH-MENT	UO2 FUEL FABRI-CATION	MOX FUEL FABRI-CATION	REACTOR	REPROCES-SING	TRANSPOR-TATION	WASTE MANAGE-MENT	SPENT FUEL STORAGE	TOTAL
PLANT WASTE GENERATED (CUBIC METERS)												
CHEMICAL COMPOUNDS	- -	- -	1 8E+05	6 9E+02	1 5E+05	0 0	- -	0 0	- -	- -	- -	3 2E+05
MILL TAILINGS	- -	7 8E+08	- -	- -	- -	0 0	- -	0 0	- -	- -	- -	7 8E+08
TRANS-U SOLIDS	- -	- -	- -	- -	- -	0 0	- -	0 0	- -	- -	- -	- -
HIGHLEVEL SOLIDS	- -	- -	- -	- -	- -	0 0	- -	0 0	- -	- -	5 5E+04	5.5E+04
OTHER RAD SOLIDS	- -	- -	8 8E+04	6 5E+04	- -	0 0	3 8E+06	0 0	- -	- -	3 4E+03	4 0E+06
PERSON-REM COMMITMENT - OCCUPATIONAL												
TOTAL BODY	1 2E+06	5 6E+05	4 4E+03	3 4E+03	5 1E+04	0 0	2 3E+06	0 0	5 4E+03	3 0E+03	1 1E+04	4 1E+06
G I TRACT	1 2E+06	2 1E+05	2 7E+03	1 7E+03	5 0E+04	0 0	2 3E+06	0 0	5 4E+03	3 0E+03	1 1E+04	3 8E+06
BONE	1 7E+06	2 3E+06	5 0E+04	3 1E+04	6 6E+04	0 0	2 3E+06	0 0	5 4E+03	2 0E+04	1 1E+04	6 5E+06
LIVER	1 2E+06	2 1E+05	5 2E+03	3 4E+03	5 0E+04	0 0	2 3E+06	0 0	5 4E+03	4 5E+03	1 1E+04	3 8E+06
KIDNEY	1 7E+06	2 6E+05	1 3E+04	8 5E+03	5 4E+04	0 0	2 3E+06	0 0	5 4E+03	4 1E+03	1 1E+04	4 4E+06
THYROID	1 2E+06	2 1E+05	4 4E+03	2 4E+03	5 0E+04	0 0	2 2E+06	0 0	5 4E+03	3 0E+03	1 1E+04	3 8E+06
LUNG	6 5E+06	4 8E+06	2 5E+04	7 5E+04	2 0E+06	0 0	2 3E+06	0 0	5 4E+03	2 5E+03	1 1E+04	1 6E+07
SKIN	1 2E+06	2 1E+05	1 3E+04	1 1E+04	5 0E+04	0 0	2 3E+06	0 0	5 4E+03	2 5E+03	1 1E+04	3 8E+06
PERSON-REM COMMITMENT - OFF-SITE U S POPULATION												
TOTAL BODY	3 0E+06	5 8E+05	4 2E+04	7 6E+01	2 5E+02	0 0	3 1E+05	0 0	1 5E+02	2.6E+00	2 8E+01	3 9E+06
G I TRACT	1 2E+05	2 2E+04	5 6E+03	6 6E+01	2 7E+02	0 0	3 0E+05	0 0	1 5E+02	5.9E-01	2 8E+01	4 5E+05
BONE	9 7E+06	1 9E+06	1 0E+05	9 2E+02	4 1E+04	0 0	1 1E+06	0 0	1 5E+02	1.3E+01	2 8E+01	1 7E+07
LIVER	2 4E+06	4 5E+05	7 2E+02	2 0E+01	5 2E+00	0 0	1 1E+05	0 0	1 5E+03	2.6E+00	2 8E+01	3 2E+06
KIDNEY	1 1E+07	2 1E+06	1 2E+04	5 2E+02	6 5E+02	0 0	3 0E+05	0 0	1 5E+03	4 0E+00	2 8E+01	1 4E+07
THYROID	7 3E+02	1 5E+02	4 8E+01	1 8E+01	4 6E+00	0 0	4 8E+05	0 0	1 5E+02	3 1E-01	2 8E+01	4 9E+05
LUNG	9 1E+05	1 8E+05	9 7E+02	4 8E+02	1 4E+02	0 0	3 0E+05	0 0	1 5E+02	1 1E+00	6 1E+01	1 4E+06
SKIN	7 3E+02	1 5E+02	2 7E+01	1 8E+01	4 6E+00	0 0	3 1E+05	0 0	1 5E+02	3.1E-01	2 4E+02	3 3E+05
PERSON-REM COMMITMENT - TO FOREIGN POPULATION FROM U S INDUSTRY												
TOTAL BODY	- -	- -	- -	- -	- -	0 0	2 1E+05	0 0	- -	- -	1 3E+02	2 1E+05
G I TRACT	- -	- -	- -	- -	- -	0 0	2 1E+05	0 0	- -	- -	1 3E+02	2 1E+05
BONE	- -	- -	- -	- -	- -	0 0	8E+06	0 0	- -	- -	1 3E+02	1 0E+06
LIVER	- -	- -	- -	- -	- -	0 0	2 1E+05	0 0	- -	- -	1 3E+02	2 1E+05
KIDNEY	- -	- -	- -	- -	- -	0 0	2 1E+05	0 0	- -	- -	1 3E+02	2 1E+05
THYROID	- -	- -	- -	- -	- -	0 0	2 1E+05	0 0	- -	- -	1 3E+02	2 1E+05
LUNG	- -	- -	- -	- -	- -	0 0	1E+05	0 0	- -	- -	2 8E+02	2 1E+05
SKIN	- -	- -	-	- -	- -	0 0	2 5E+05	0 0	- -	- -	1 1E+04	2 6E+05

TABLE S-3 From WASH 1248

Summary of Environmental Considerations
for Uranium Fuel Cycle

(Normalized to Model LWR Annual Fuel Requirement)

	Total	Maximum Effect per Annual Fuel Requirement of Model 1000 MWe LWR
Natural Resource Use		
Land (acres)		
Temporarily committed	63	
Undisturbed area	45	
Disturbed area	18	Equivalent to 90 MWe coal-fired power plant
Permanently committed	4.6	
Overburden moved (millions of MT)	2.7	Equivalent to 90 MWe coal-fired power plant
Water (millions of gallons)		
Discharged to air	156	∿ 2% model 1000 MWe LWR with cooling tower
Discharged to water bodies	11,040	
Discharged to ground	123	
Total	11,319	< 4% of model 1000 MWe LWR with once-through cooling
Fossil Fuel		
Electrical energy (thousands of MW-hr.)	317	< 5% of model 1000 MWe LWR output
Equivalent coal (thousands of MT)	115	Equivalent to the consumption of a 45 MWe coal-fired power plant
Natural gas (millions of scf)	92	< 0.2% of model 1000 MWe energy output
Effluents - Chemical (MT)		
[1] Gases (including entrainment)		
[2] SO_x	4,400	
NO_x	1,177	Equivalent to emissions from 45 MWe coal-fired plant for a year.
Hydrocarbons	13.5	
CO	28.7	
Particulates	1,156	
Other Gases		
F^-	0.72	Principally from UF_6 production enrichment and reprocessing. Concentration within range of state standards - below level that has effects on human health.

[1] Estimated effuents based upon combustion of equivalent coal for power generation.

[2] 1.2% from natural gas use and process.

TABLE S-3 (cont.) From WASH 1248

	Total	Maximum Effect per Annual Fuel Requirement of Model 1000 MWe LWR
Effluents - Chemical (MT) (cont.)		
Liquids		
$SO_4^=$	10.3	From enrichment, fuel fabrication, and reprocessing steps. Components that constitute a potential for adverse environmental effect are present in dilute concentrations and receive additional dilution by receiving bodies of water to levels below permissible standards. The constituents that require dilution and the flow of dilution water are:
NO_3^-	26.7	
Fluoride	12.9	
Ca^{++}	5.4	
Cl^-	8.6	
Na^+	16.9	
NH_3	11.5	NH_3 - 600 cfs
Fe^3	0.4	NO_3 - 20 cfs
		fluoride - 70 cfs
Tailings Solutions (thousands of MT)	240	From mills only - no significant effluents to environment.
Solids	91,000	Principally from mills - no significant effluents to environment.
Effluents - Radiological (curies)		
Gases (including entrainment)		
Rn-222	75	Principally from mills - Maximum annual dose rate < 4% of average natural background within 5 miles of mill. Results in 0.06 man-rem per annual fuel requirement.
Ra-226	0.02	
Th-230	0.02	
Uranium	.032	
Tritium (thousand)	16.7	Principally from fuel reprocessing plants - Whole body dose is 4.4 man-rem per annual fuel requirements for population within 50-mile radius. This is < 0.005% of average natural background dose to this population. Release from Federal Waste Repository of 0.005 Ci/yr has been included in fission products and transuranics total.
Kr-85 (thousands)	350	
I-129	0.0024	
I-131	0.024	
Fission Products & Transuranics	1.01	
Liquids		
Uranium & daughters	2.1	Principally from milling - included in tailings liquor and returned to ground - no effluents; therefore, no effect on environment.
Ra-226	.0034	From UF_6 production-concentration 5% of 10 CFR 20 for total processing of 27.5^6 model LWR annual fuel requirements.
Th-230	.0015	

TABLE S-3 (cont.) From WASH 1248

	Total	Maximum Effect per Annual Fuel Requirement of Model 1000 MWe LWR
Effluents - Radiological (curies) (cont.)		
Liquids (cont.)		
Th-234	0.01	From fuel fabrication plants - concentration 10% of 10 CFR 20 for total processing 26 annual fuel requirements for model LWR.
Ru-106	0.15*	From reprocessing plants - maximum concentration 4% of 10 CFR 20 for total reprocessing of 26 annual fuel requirements for model LWR.
Tritium (thousands)	2.5	
Solids (buried)		
Other than high level	601	All expect 1 Ci comes from mills - included in tailings returned to ground - no significant effluent to the environment, 1 Ci from conversion and fuel fabrication is buried.
Thermal (billions)	3,360	< 7% of model 1000 MWe LWR.
Transportation (man-rem)		
Exposure of workers and general public	0.334	

* Cs-137 (.075 Ci/AFR) and Sr-90 (.004 Ci/AFR) are also emitted.