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# GENERIC WASTE MANAGEMENT CONCEPTS FOR SIX LWR FUEL CYCLES

MASTER

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**April 1979**

**DIVISION OF ENVIRONMENTAL IMPACT STUDIES**
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This report was originally commissioned to supplement the treatment of waste management issues provided in the Generic Environmental Statement on the Use of Recycle Plutonium in Mixed Oxide Fuel in Light Water Cooled Reactors (GESMO) (NUREG-0002). On December 27, 1977, the Commission terminated the GESMO proceeding. Although no longer relevant to the GESMO proceeding, this study report is being issued for informational purposes only and does not necessarily represent the views of the Nuclear Regulatory Commission staff. No inferences should be drawn from it vis-a-vis the GESMO, the GESMO proceeding or the plutonium recycle decisionmaking process.

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## ABSTRACT

This report supplements the treatment of waste management issues provided in the Generic Environmental Statement on the use of recycle plutonium in mixed oxide fuel in light water cooled reactors (GESMO, NUREG-0002). Three recycle and three no-recycle options are described in this document. Management of the radioactive wastes that would result from implementation of either type of fuel cycle alternative is discussed. For five of the six options, wastes would be placed in deep geologic salt repositories for which thermal criteria are considered. Radiation doses to the workers at the repositories and to the general population are discussed. The report also covers the waste management schedule, the land and salt commitments, and the economic costs for the management of wastes generated.

# 1. INTRODUCTION

Light water nuclear reactors are currently fueled with slightly enriched uranium. While the reactor operates, some of the uranium is converted to plutonium, which fissions in place, providing about one-third of the reactor's total power output over the useful life of the fuel. Fuel burnup also creates other byproducts, which gradually impede the nuclear reaction, even though substantial quantities of fissile uranium and plutonium still remain in the fuel. When the useful life of the fuel is over, the remaining fissile uranium and plutonium can be separated from the other materials in the spent fuel, converted into uranium and plutonium oxides, and recycled into the reactor as fuel. The process of extracting and reusing the elements in this fashion is known as "full recycle," and fuel containing recycled plutonium is termed "mixed oxide" fuel. The extraction itself is known as fuel reprocessing. In the "Purex" process, which has been used successfully for many years, the spent fuel rods are first chopped up and the fuel is dissolved in nitric acid and separated from the insoluble cladding. Then by a series of extraction processes the uranium and plutonium are first separated from the nitric acid solution and then from each other. The remaining solution contains high-activity waste, the fission products, and the long-lived alpha emitters--the actinides.

The U.S. Nuclear Regulatory Commission (NRC) and its predecessor, the U.S. Atomic Energy Commission (AEC), determined that widescale recovery and recycle of plutonium fuel in light water cooled nuclear power reactors warranted analysis apart from that given for the licensing of any single recycle facility, and that adoption of rules governing such widescale use would constitute a major Federal action which would have the potential to significantly affect the quality of the human environment. Accordingly, pursuant to the National Environmental Policy Act of 1969 (NEPA), Section 102(2)(C), NRC has prepared a final Generic Environmental Statement on the Use of Recycle Plutonium in Mixed Oxide Fuel in Light Water Cooled Reactors (GESMO)[1] to assess the impacts of the implementation of plutonium recycle.

In reviewing the GESMO document, the hearing board established by the NRC found deficiencies in those sections concerning the proposed management of wastes from the back-end of the various fuel cycles addressed. The report presented here, originally commissioned to supplement that treatment of waste management issues provided in the GESMO Statement and in the GESMO proceeding, was prepared in 1977 and reflects the information available at that time. Obviously, new information on costs and technology is now available. However, no attempt has been made to incorporate this new information.

## 1.1 SCOPE

The initial scope of this study was mandated by the GESMO hearing board. The board's criticisms regarding the waste management sections of GESMO can be grouped into four general categories:

1. More fuel cycle options should have been considered;
2. More detail should have been given in the descriptions of the options and their environmental effects;
3. There was not enough contrast between the options cited; and
4. The post-2000 waste management scenarios were not described.

To address the first criticism, two no-recycle options were added to the no-recycle deep geologic storage option described in GESMO, and another recycle option was added to the two recycle options covered in GESMO. These additions brought to six the total number of options addressed. [Thus, for the purposes of this document, two basic alternatives (recycle and no-recycle) are involved, and for each alternative, three possible courses of action or "options" are considered.]

To address the second criticism, more detail was added under topics such as waste amounts and packaging, facility descriptions, procedures, radioactive releases, doses, and natural resource commitments for the various options.

The third criticism was addressed throughout the document by explication of the differences in all major areas between the options.

To address the fourth criticism, it was assumed that at the year 2000, the nuclear industry would have grown to 507 gigawatts-electric (GWe) of capacity and that after the year 2000, no new reactors would be built. Each reactor was assumed to have a lifetime of 30 years. All waste generated up to the year 2030, when the last reactor is assumed to be shut down, was accounted for, thus giving the total waste management picture for each option. The wastes are assumed to be disposed in Federal repositories.

The tasks defined by the scope of this document are carried out in the following sections. In the remainder of Section 1, the six options considered are delineated and the assumptions upon which this document is based are presented. The wastes and the facilities and procedures to be used at the waste repositories for the options considered are described in Section 2. The thermal analysis used to calculate the spacing of certain waste types in the underground salt repositories is described in Section 3. In Section 4, dose calculations and radioactive releases for normal operations and accidents at the repositories are presented for each of the six options. The nuclear power generation schedule and waste inventory for the years 1960-2040 are detailed in Section 5. Natural resource commitments for each option are given in Section 6. Economic considerations for the various waste management options are given in Section 7. Four appendices follow the main body of the text. The isotopic characteristics of various waste types are given in Appendix A. Some of the major properties of the calcined high-level solidified waste are given in Appendix B. The estimated quantities of waste to be handled for the various fuel cycle options, as calculated through a computer program, are given in Appendix C. Some geological requirements for underground disposal of nuclear wastes are described in Appendix D.

## 1.2 DELINEATION OF OPTIONS

Three recycle and three no-recycle options are considered in this document. The recycle options are:

1. Recycle of uranium only, with the plutonium stored below ground in a retrievable form for possible future use as an energy resource;
2. Recycle of uranium only, with the plutonium considered to be a waste material; and
3. Full recycle of both uranium and plutonium.

The no-recycle options are:

1. Surface storage of spent fuel;
2. Deep geologic emplacement of spent fuel so that it could be retrieved at some future date (stowaway option); and
3. Deep geologic emplacement of spent fuel with no intent or designed features of retrievability (throwaway option).

Surface storage of spent fuel would be only an interim solution. It eventually would be necessary to dispose this spent fuel, either by reprocessing and burying the resultant wastes or by burying the intact spent fuel assemblies. The final disposition of the spent fuel assemblies following surface storage is not considered in this document. The other five options involve emplacement of the wastes in deep geologic salt formations. Because of repository design similarities, the two retrievability options (plutonium and spent fuel) can be easily converted to the respective non-retrievability modes of operation.

## 1.3 ASSUMPTIONS

The nuclear power industry growth assumed for this document is based upon the Energy Research and Development Administration (ERDA) mid-1976 forecast of 507 GWe in the year 2000. No new reactors are assumed to come online after that year. The reactors are assumed to produce power in a ratio of 2 PWR:1 BWR. It is assumed that a PWR (pressurized water reactor) fuel assembly is charged with 0.45 metric ton (MT) [0.50 short ton (ST)] of fuel, and a BWR (boiling water reactor) assembly is charged with 0.20 MT (0.22 ST).

Every reactor is assumed to have a 30-year operational lifetime, and all reactors are assumed to have a fuel burnup of 33 gigawatt-days per metric ton of heavy metal (GW-days/MTHM)* and a thermal efficiency of 32.7%. The maximum plant capacity factor attained over the life of the reactor is assumed to be 0.8, and the average is assumed to be slightly lower. This would result in an annual discharge rate of about 26 MT (29 ST) of fuel per GWe. The number of BWR and PWR bundles discharged each year can be calculated from the fuel discharge rate, the reactor type ratio, and the bundle weights. For each GWe, 38.5 PWR and 43.3 BWR assemblies would be discharged annually. This is a spent fuel assembly ratio of 1 PWR:1.125 BWR.

*Heavy metal refers to the total actinides charged to the reactor.

The fuel-loading model used in this report is shown in Table 1.1. It is assumed that fuel is not discharged from a reactor during the first two years of operation, nor is the reactor reloaded during the last two years of operation. A double discharge occurs at the time of reactor shutdown to account for the extra fuel left in the core.

All wastes are assumed to have been out of the reactor for at least ten years before arriving at the repository. This ten-year out-of-reactor time could involve ten years of storage as unreprocessed spent fuel, or any time combination of spent fuel storage and post-reprocessing storage as reprocessing wastes (e.g., five years storage as unreprocessed spent fuel, followed by reprocessing and then five years storage as reprocessing wastes).

Table 1.1. Reactor Model--Amount (MT) of Fuel per GWE

| Life, years | New Fuel | Discharged Fuel |
|---|---|---|
| 1 | 87 | 0 |
| 2 | 0 | 0 |
| 3-28 | 26 | 26 |
| 29-30 | 0 | 26 |
| 31 | - | 35 |

### 1.3.1 Recycle Options

For the reprocessing schedule for all three recycle options, the Allied General Nuclear Services (AGNS) Reprocessing Plant (Barnwell) is assumed to start operation in 1982 with a throughput capacity of 300 MTHM/yr. The capacity is assumed to be increased by 300 MTHM/yr until the final operating capacity of 1500 MTHM/yr is achieved. A second reprocessing plant is assumed to start operation in 1986 with an initial capacity of 500 MTHM/yr, increasing by 500 MTHM/yr until the final operating capacity of 3000 MTHM/yr is reached. A third reprocessing plant is assumed to begin operation in 1991, with the same capacity and staging as the second facility. All fuel for any of the three recycle options is assumed to be reprocessed after being out of the reactor core at least 160 days. The plutonium and all wastes generated for any of the recycle options are assumed to be out of the reactor at least ten years before final disposition.

#### 1.3.1.1 Uraniun-Only Recycle

The basic assumptions for the uranium-only (U-only) recycle options are:

1. Spent fuel is reprocessed as soon as reprocessing plant capacity becomes available;
2. The oldest spent fuel is reprocessed first;
3. Reprocessing continues at full capacity until the year 2030;
4. All wastes are shipped to the Federal repository when they are at least ten years old (age is based on years after discharge from the reactor);
5. The backlog of spent fuel not reprocessed by the year 2030 is considered waste and will be disposed when ten years old.

#### 1.3.1.2 Full Recycle

The basic assumptions for the full recycle option, referred to as mixed oxide (MOX)* reprocessing are:

1. The MOX fuel is as defined in GESMO for a 1.15 SGR (self-generating reactor).**

*Mixed oxide, or MOX, refers to fresh reactor fuel consisting of a combination of plutonium dioxide and uranium dioxide.

**A self-generating reactor is an equilibrium condition in which the amount of plutonium recovered from reprocessing MOX and $UO_2$-only fuel rods is equal to the amount of plutonium in the MOX fuel rods originally loaded into the reactor. A 1.15 SGR is one which requires 15% more plutonium from other sources in addition to that recovered from reprocessing the spent fuel to be at equilibrium (see Ref. 1).

2. Three generations of MOX fuels are used. The GESMO analysis indicated that after three recycles in an LWR, the MOX fuel would have built up enough neutron-absorbing isotopes to require uneconomic uranium enrichment to maintain reactivity. MOX wastes also have different decay characteristics and isotopic compositions than $UO_2$ fuels.
3. All fuels undergo a four-year BWR reactor cycle. This assumption is used because the maximum amount of time between full-core replacement is four years.
4. MOX fuels are reprocessed about 160 days after removal from a core.
5. The total turn-around time between MOX generations is two years. This allows time for reprocessing, fabrication, and shipment.
6. A reactor can start on MOX fuels if it is between three and ten years old and it continues on MOX fuels until shutdown. The initial time is based on assumptions regarding stabilization of reactor systems, and the final age is based on the total MOX-fuel-stabilization time of 16 years.
7. The reprocessing schedule is determined by demand for fuel for those reactors old enough to begin using MOX fuel. No more fuel is reprocessed than can be used. The demand will dictate the rate of decrease in reprocessing as the reactors shut down due to age.
8. Reprocessing is based upon a priority system. Higher generation MOX spent fuel (MOX 2 and MOX 1) are reprocessed first, followed by $UO_2$ spent fuel. The total reprocessing amounts are kept within the constraints of assumption 7.
9. A maximum of 40% of the core can be MOX fuel, with the remaining 60% being enriched $UO_2$ fuel. The MOX fuel is assumed to have a maximum plutonium content of 4.5%, giving a maximum core average of 1.8% plutonium.
10. The amount of spent fuel that will be reprocessed will be that required to meet the projected needs of the nuclear power industry. Because of these projections, there will be spent fuel which will not be reprocessed. This spent fuel will be treated as waste and will be disposed when it is ten years old.

### 1.3.2 No-Recycle Options

Once-through spent nuclear fuel, still in reactor assemblies, will be the major waste for all three no-recycle options. The discharge rate of spent fuel is governed by the burnup assumptions, and the number of spent fuel assemblies is governed by the discharge rate and the PWR to BWR power ratio (two PWRs for each BWR).

### Reference

1. "Final Generic Environmental Statement on the Use of Recycle Plutonium in Mixed Oxide Fuel in Light Water Cooled Reactors" (GESMO), U.S. Nuclear Regulatory Commission, NUREG-0002, August 1976.

# 2. DESCRIPTION AND MANAGEMENT OF WASTES

Management of the radioactive wastes that would result from implementation of the two general types of fuel cycle alternatives--recycle or no-recycle--is discussed in this section. The wastes are described and means for their final disposition are discussed.

## 2.1 DESCRIPTION OF WASTES

For the purposes of this discussion, the wastes of concern are the spent nuclear fuel and the various types of radioactive materials that would result from the reprocessing of nuclear fuel under the three recycle options. The reprocessing wastes generally can be grouped into six categories: high-level solidified waste (HLSW), fuel bundle residues (hulls), plutonium dioxide ($PuO_2$) spiked with fission products, transuranic intermediate-level waste (TRU-ILW), transuranic low-level waste (TRU-LLW), and nontransuranic low-level waste (non-TRU).*

Under the three no-recycle options, the spent fuel would be left in the fuel assemblies removed from a reactor, and thus only one type of waste--spent fuel--would have to be dealt with. Under the various recycle options, however, all types of wastes mentioned above, including spent fuel,** would be present and would have to be considered in any waste management program. A summary of the types of wastes that would be handled for each of the six options considered is given in Table 2.1.

Table 2.1. Types of Wastes for the Six Fuel Cycle Options

| | Waste Type | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| Fuel Cycle Option | SF ($UO_2$) | SF (MOX) | HLSW ($UO_2$) | HLSW (MOX) | Spiked $PuO_2$ | Hulls | TRU-ILW | TRU-LLW |
| No-recycle, surface storage | X | | | | | | | |
| No-recycle, deep geologic stowaway | X | | | | | | | |
| No-recycle, deep geologic throwaway | X | | | | | | | |
| U-recycle, Pu stored | X | | X | | X | X | X | X |
| U-recycle, Pu disposed | X | | X | | X | X | X | X |
| Full recycle, deep geologic repositing | X | X | X | X | | X | X | X |

*The non-TRU low-level wastes generally have been routinely buried in various commercial landfill-type operations. Such operations are not considered in this report.

**Because of scheduling and reprocessing capacity, there would be some spent fuel that would not be reprocessed under all the recycle options (see Sec. 5).

### 2.1.1 Spent Fuel

Upon discharge from a reactor, the intact fuel assemblies are radioactive because of fission products and activation products formed during reactor operation. The level of radioactivity and the final composition of the spent fuel are directly related to the type of fuel charged to the reactor, the length of time that the assemblies were in the reactor, and the reactor power level. Isotopic mixes of spent fuel ten years and 160 days out of the core for the U-only recycle and the no-recycle options are shown in Table A.2 of Appendix A.

For the full recycle option, the reactors would be gradually charged with a greater percentage of recycled plutonium until the 1.15 SGR equilibrium level was achieved. The spent fuel and subsequent wastes from the 1.15 SGR equilibrium level would be more radioactive and give off more heat than the wastes from pre-equilibrium levels of operation. Isotopic mixes for spent fuel from this equilibrium level ten years plus 160 days out of core are given in Table A.5 of Appendix A.

### 2.1.2 High-Level Waste

High-level waste is defined as the raffinate from the first solvent extraction step at a reprocessing plant.[1] In practice, additional liquid wastes resulting from further reprocessing of the spent fuel could be merged with this high-level liquid waste and the resultant mixture still would be called high-level waste.[2] For all three recycle cases, this waste stream is assumed to contain 0.5% of both the uranium and plutonium and 98.5% of the fission products and other actinides that were originally in the spent fuel. The remaining 1.5% of the fission products would be left in the plutonium for safeguards reasons. This would produce a spiked $PuO_2$ mixture that by weight would be 95% $PuO_2$ and 5% fission products. In the reprocessing operation, the volatile fission product gases xenon and krypton would be released, as would most of the iodine, bromine, and tritium (H-3).

It currently is generally considered that before ultimate disposal, liquid high-level wastes should be solidified so as to reduce their potential for environmental impact and to increase the ease and safety of handling. Therefore, in the rest of this report these wastes are assumed to be in a solidified form and are referred to as high-level solidified wastes (HLSW) or simply as high-level wastes (HLW).

Several methods for solidifying high-level liquid wastes have been proposed and studied. For this report, use of a fluidized bed calcination process is assumed. This calcined HLSW is assumed to be left as a powder rather than being put into a glass or metal matrix, as has been considered in some studies. This is a conservative assumption for the analyzing of the occupational and accident doses from the handling of HLSW because this powder is easily dispersable and highly respirable. Isotopic mixtures of this calcine are given in Tables A.1 and A.4 in Appendix A, and some of its major properties are given in Appendix B. It is assumed that the HLSW would be packaged in stainless steel canisters prior to disposal.

### 2.1.3 Hulls

Zircaloy cladding, stainless steel and Inconel support rods, neutron absorbing rods, end fittings, springs, and spacer elements would be left after the spent fuel pellets were dissolved in the first nitric acid solution at the reprocessing plant. These wastes are collectively referred to as hulls. The hulls are assumed to be uncompacted and packaged in canisters of the same design as the HLSW canisters. Although the hulls would be leached in a nitric acid solution, they are assumed to retain 0.1% of the spent fuel isotopes. Because of the activation of the hulls along with this residual spent fuel, shielding of the hull canisters would be required.

### 2.1.4 Transuranic Intermediate-Level Waste (TRU-ILW)

TRU-ILW are transuranic wastes which require shielding for protection from the emitted radiation. The TRU-ILW would come mainly from the reprocessing facility and consist of contaminated ion-exchange resins, filters, clothes, rubber gloves, tools, glassware, and similar items. It is assumed that these wastes would be neither compacted nor incinerated. The TRU-ILW would be packaged in containers similar to the HLSW canisters. (The HLSW, hulls, and TRU-ILW types of waste are referred to as "canistered" wastes.)

### 2.1.5 Transuranic Low-Level Waste (TRU-LLW)

The TRU-LLW consists of TRU wastes that do not require shielding. These also would be generated at the reprocessing facility. The TRU-LLW is assumed to be packaged in 55-gallon drums without preliminary compaction or incineration.

### 2.1.6 Plutonium Dioxide ($PuO_2$)

It is assumed that plutonium recovered from reprocessing of spent fuel would be converted to the oxide $PuO_2$ at the reprocessing plant since pursuant to 10 CFR 70.42,[3] the Federal Government requires that plutonium in excess of 20 curies per package be shipped as a solid. Plutonium from $UO_2$ fuel, at 33 GWd/MTU burnup, has a specific activity of about 0.5 curies per gram (alpha). It is further assumed that $PuO_2$ would be shipped and stored in containers holding about 6 kg (13 lb) of $PuO_2$, a quantity which would present no criticality hazard in a suitably designed container.

In GESMO it was assumed that sufficient fission products would be left in the plutonium to produce a radiation level that would discourage theft or diversion for malevolent purposes. This could be achieved by "spiking" the $PuO_2$ with a small part of the high-level waste (5% fission products by weight). This spiked $PuO_2$ is assumed to be placed in thin-walled canisters 10 cm (4 inches) in diameter by 61 cm (2 ft) long.[4] The canisters are assumed to be sealed in overpacks similar in size and shape to 55-gallon drums and then stored in the repository.[4,5] The isotopic mix of this spiked plutonium is shown in Table A.3 of Appendix A.

Some of the important characteristics of these types of wastes for uranium and MOX fuel reprocessing are shown in Table 2.2.

## 2.2 WASTE MANAGEMENT

In the recycle options, the recovered uranium would be recycled to a fuel fabrication plant, and the recovered plutonium would be either recycled with the uranium (full recycle) or stored in a retrievable mode or disposed (U-only recycle). (Schematic diagrams of the full recycle, U-only recycle, and no-recycle options are shown in Figs. 2.1 through 2.3.) It is assumed that the spent fuel and all reprocessing wastes except the non-TRU low-level wastes would be sent to Federal repositories for storage or disposal.

Only a relatively brief description of model repositories is given here. More detailed information concerning proposed repositories can be found in other documents, such as Reference 6. Two types of Federal repositories are modeled for this study: one for reprocessing wastes and one for unreprocessed spent fuel. Repositories for unreprocessed spent fuel would be needed for recycle and no-recycle options. Repositories for reprocessing wastes would be required only for the recycle options.

### 2.2.1 Reprocessing Wastes Disposal

A flow diagram for a reprocessing wastes repository is shown in Figure 2.4, and a schematic drawing of such a repository is shown in Figure 2.5. Plutonium storage/disposal facilities are assumed to be added if the full recycle option is not chosen. The Federal repository is assumed to be in a rock salt formation. Locations of rock salt deposits in the United States are shown in Figure 2.6. A secured area of approximately 80 hectares (ha) (200 acres) would contain the various aboveground facilities for operation of the model repository. An underground storage area with a floor area of about 800 ha (2000 acres) would be excavated for the burial of the nuclear wastes. A safety buffer zone of an additional 1200 ha (3000 acres) would be established. No underground activity would be permitted within this 2000-ha (5000-acre) area; however, some restricted surface activity might be allowed.

The model Federal repository is described below in terms of procedures and facilities for the handling and storage of three types of wastes: (1) canistered wastes (HLSW, hulls, TRU-ILW), (2) TRU-LLW, and (3) spiked $PuO_2$.

#### 2.2.1.1 Canistered-Waste Facility

A canistered-waste building on the surface would house receiving, decasking, overpacking, and surge pool facilities and operations. There would be a shaft leading from this building to a mine-level receiving station through which the canistered wastes would pass enroute to emplacement in holes drilled in rooms in the salt formation. The canistered-waste building would be composed of three major areas: a cask receiving and inspection area, a pool for cask unloading and canister surge storage, and an encapsulation area. Canistered wastes would be handled and processed remotely in either air or water within shielded facilities constructed of reinforced concrete with shielding walls. All effluent air would be filtered.

Table 2.2. Characteristics of Reprocessed $UO_2$ and MOX Wastes

| Waste Form | kg/MTHM[a] | kg/m$^3$ | m$^3$/MTHM[a] | W/MTHM[a] | Waste Canister, m$^3$/can |
|---|---|---|---|---|---|
| $UO_2$ Wastes | | | | | |
| HLSW[b] | 123 | 2200 | 0.0559 | 1110 | 0.177 |
| Hulls[c] | 326 | 1000[d] | 0.326 | 28.4 | 0.177 |
| TRU-ILW[e] | 2430 | 1000[d] | 2.43 | 0.324 | 0.177 |
| TRU-LLW[f] | 972 | 1000[d] | 0.972 | 0.197 | 0.167 |
| $PuO_2$[g] | 10 | 2000 | 0.005 | 256 | 0.003 |
| MOX Wastes | | | | | |
| HLSW[b] | 124 | 2200 | 0.0565 | 2290 | 0.177 |
| Hulls[c] | 326 | 1000[d] | 0.326 | 30.0 | 0.177 |
| TRU-ILW[e] | 2430 | 1000[d] | 2.43 | 0.619 | 0.177 |
| TRU-LLW[f] | 8270 | 1000[d] | 8.27 | 1.68 | 0.167 |

[a]MTHM refers to the metric tons of heavy metal reprocessed based on the assumptions: 33 Gwd/MTHM, 30 MW/MTHM, 2/3 PWR, 1/3 BWR.

[b]100% of H-3 and noble gas fission products, and 99.9% of I and Br released; 0.5% U and Pu remain.

[c]Includes 0.1% irradiated fuel.

[d]Wastes uncompacted.

[e]Includes 0.89 grams of Pu/m$^3$ and 0.025% of the fission products.

[f]Includes 8.9 grams Pu/m$^3$.

[g]Includes 0.5 kg fission products per MTHM reprocessed.

Table based on information from:

J. O. Blomeke and C. W. Kee, "Projections of Waste to be Generated," presented at the International Symposium on the Management of Wastes from the LWR Fuel Cycle, 11-16 July 1976, Denver, Colorado, CONF-76-0701.

C. W. Kee, A. G. Croff, and J. O. Blomeke, "Updated Projections of Radioactive Wastes to be Generated by the U.S. Nuclear Power Industry," Oak Ridge National Laboratory, ORNL/TM-5427, December 1976.

"Alternatives for Managing Wastes from Reactors and Post-Fission Operations in the LWR Fuel Cycle," Volume 2, U.S. Energy Research and Development Administration, ERDA-76-43, May 1976.

"Environmental Survey of Reprocessing and Waste Management Portions of the LWR Fuel Cycle," U.S. Nuclear Regulatory Commission, NUREG-0116, October 1976.

"Final Generic Environmental Statement on the Use of Recycle Plutonium in Mixed Oxide Fuel in Light Water Cooled Reactors," Chapter IV, U.S. Nuclear Regulatory Commission, NUREG-0002, August 1976.

B. L. Cohen, "The Disposal of Radioactive Wastes from Fission Reactors," Scientific American 236(6):21-31, June 1977.

J. W. Wachter, "Effect of Fuel Recycling on Radioactivity and Thermal Power of High Level Wastes (Draft)," prepared for the U.S. Nuclear Regulatory Commission by the Oak Ridge National Laboratory, ORNL/NUREG/TM-146, December 1977.

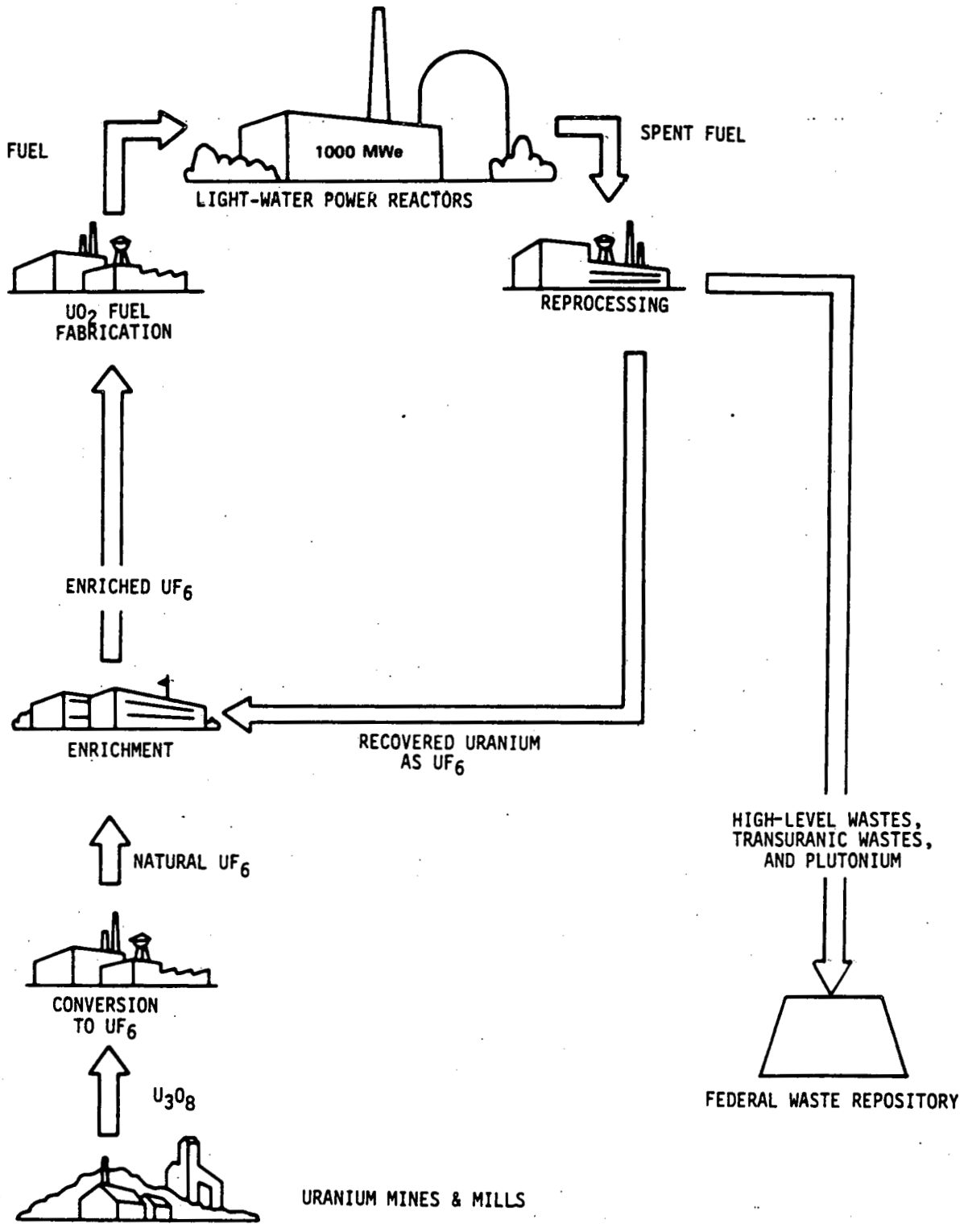



Fig. 2.1. Uranium-Only Recycle Fuel Cycle. (Fig. 3.2 in "Environmental Survey of the Reprocessing and Waste Management Portions of the LWR Fuel Cycle," U.S. Nuclear Regulatory Commission, NUREG-0116, October 1976.)

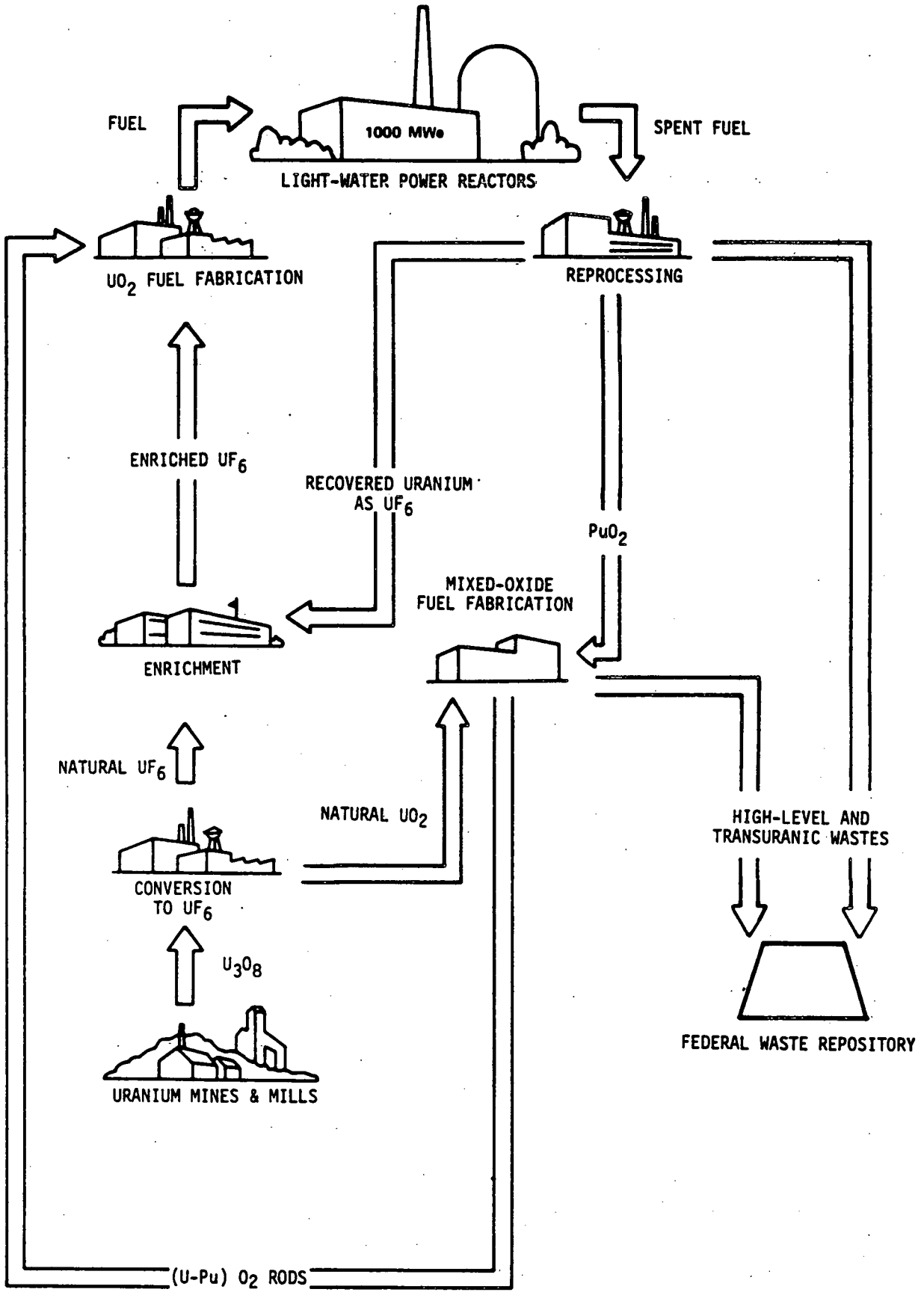



Fig. 2.2. Full Recycle Fuel Cycle. (Fig. 3.3 in "Environmental Survey of the Reprocessing and Waste Management Portions of the LWR Fuel Cycle," U.S. Nuclear Regulatory Commission, NUREG-0116, October 1976.)

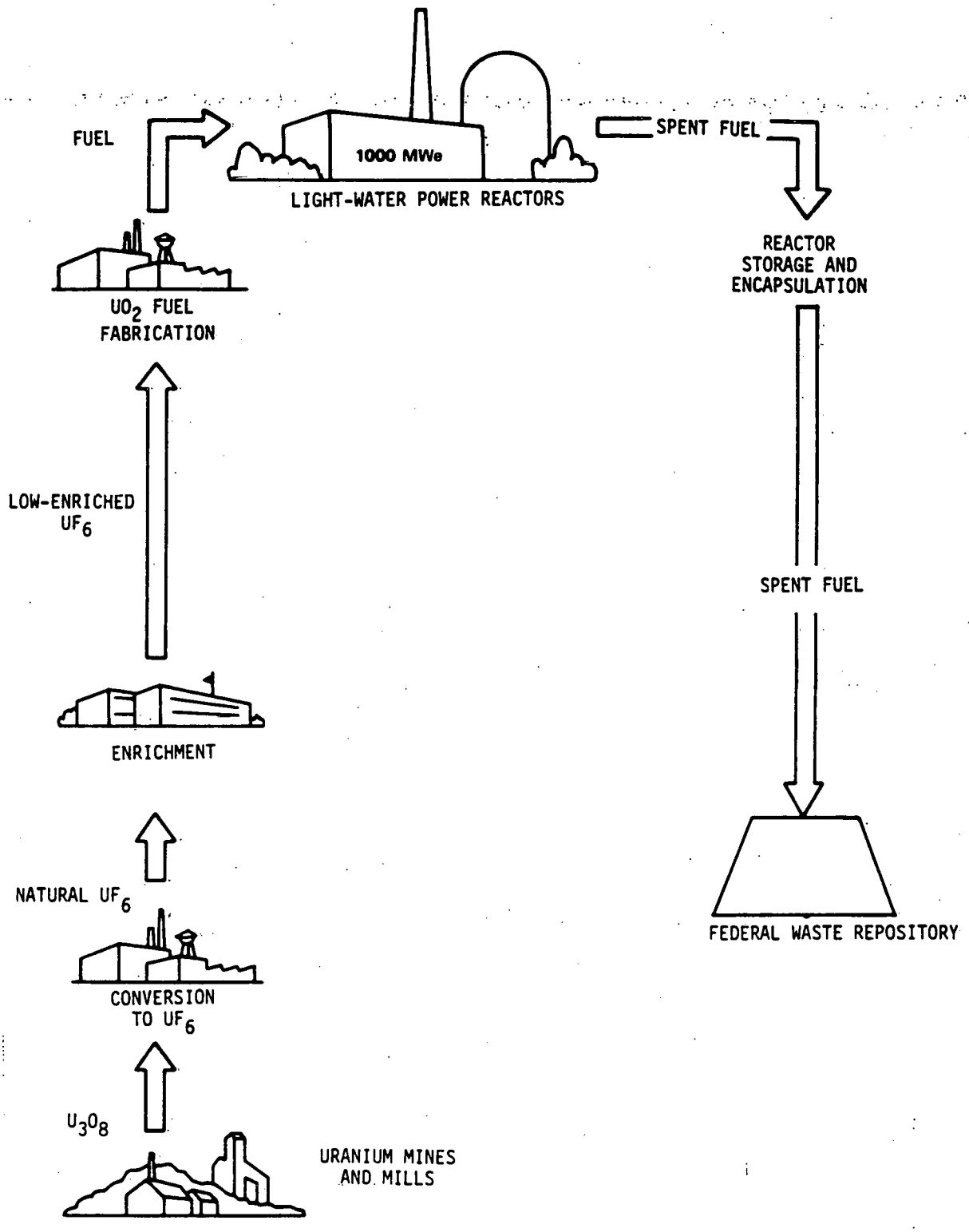
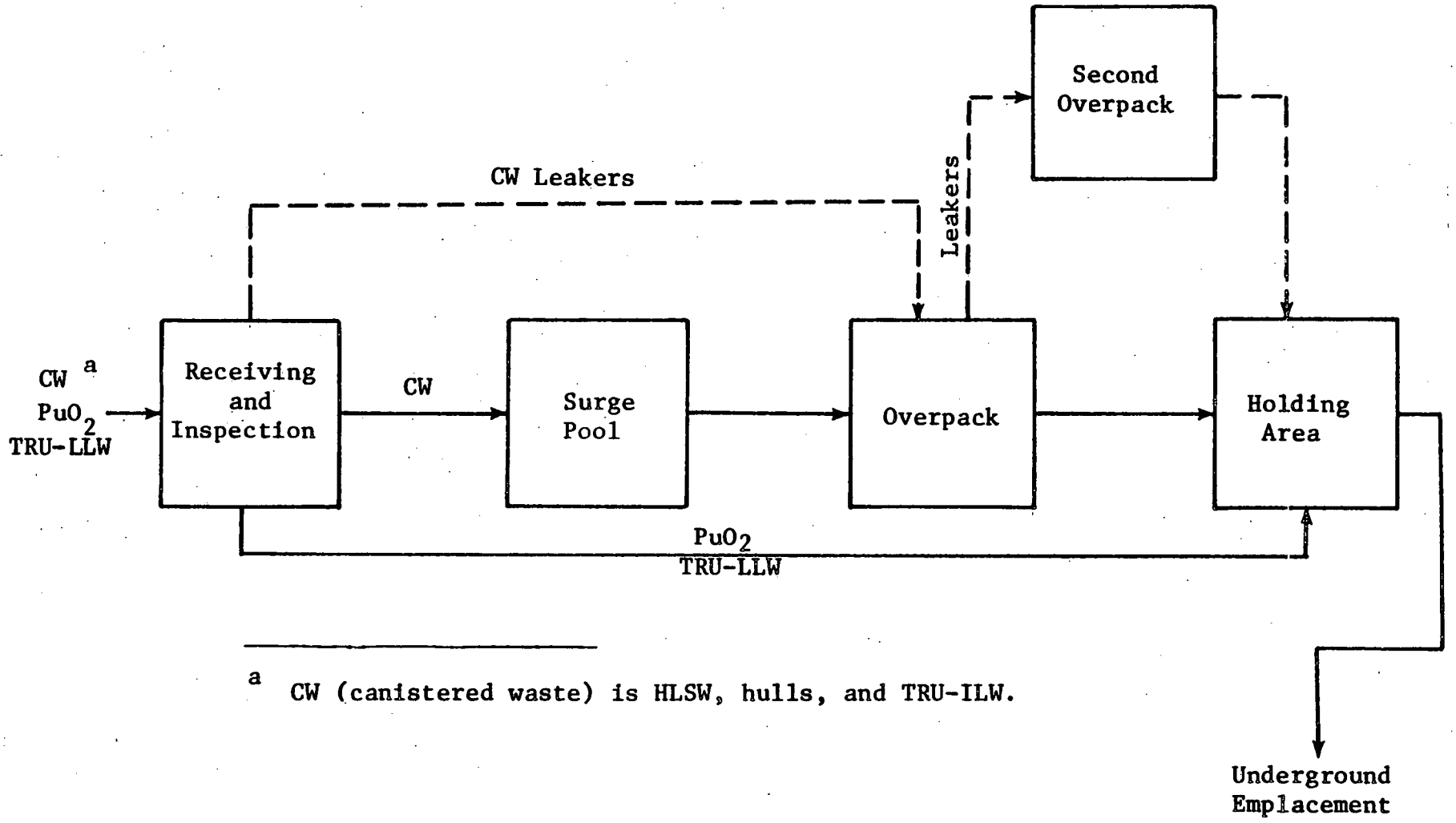



Fig. 2.3. No-Recycle Fuel Cycle. (Fig. 3.1 in "Environmental Survey of the Reprocessing and Waste Management Portions of the LWR Fuel Cycle," U.S. Nuclear Regulatory Commission, NUREG-0116, October 1976.)




[a] CW (canistered waste) is HLSW, hulls, and TRU-ILW.

Fig. 2.4. Flow Diagram for Reprocessing Wastes Repository.

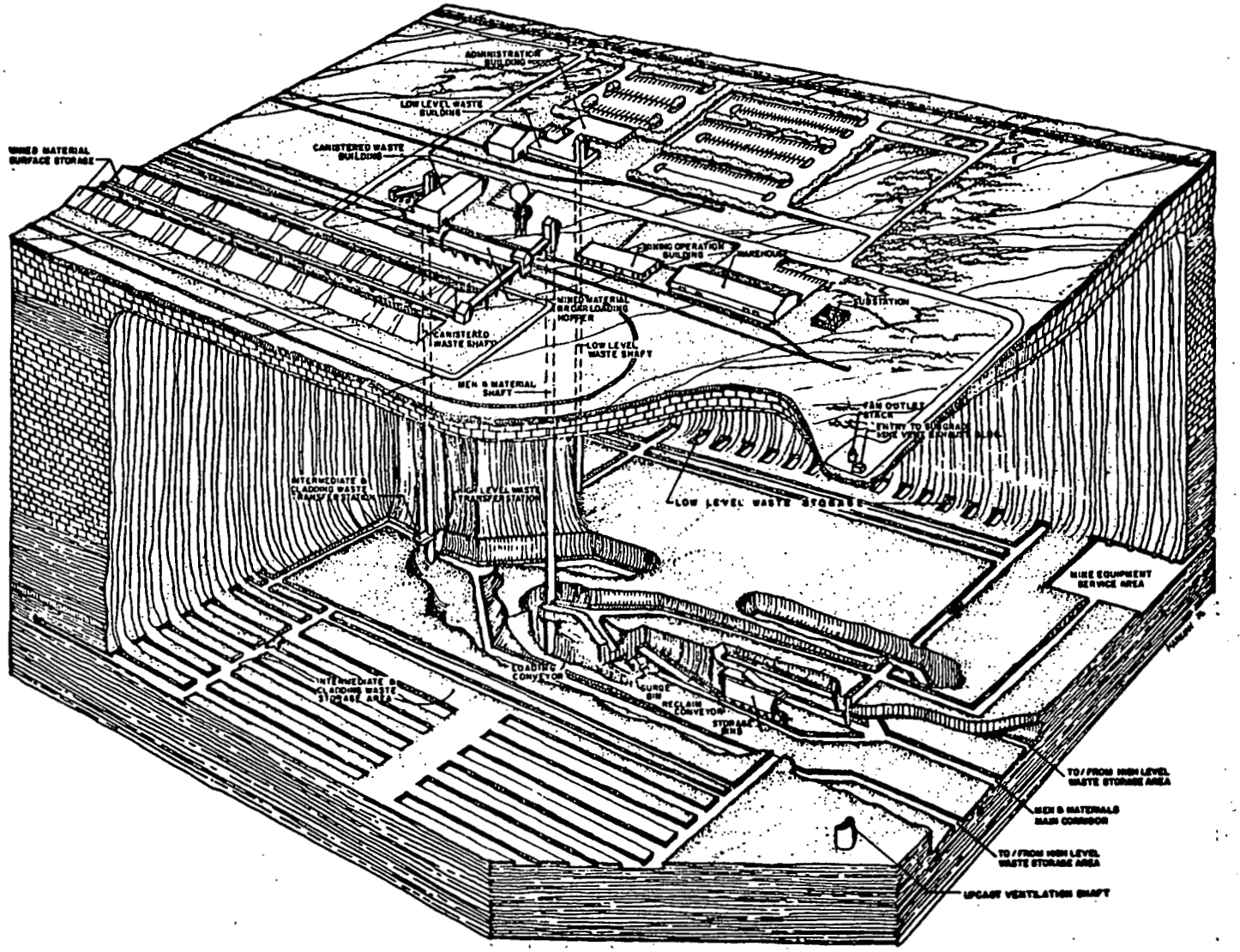



Fig. 2.5. Schematic Diagram of Waste Isolation Facility. (From "Waste Isolation Facility Description - Bedded Salt," Office of Waste Isolation, Y/OWI/SUB-76/16506, September 1976.)

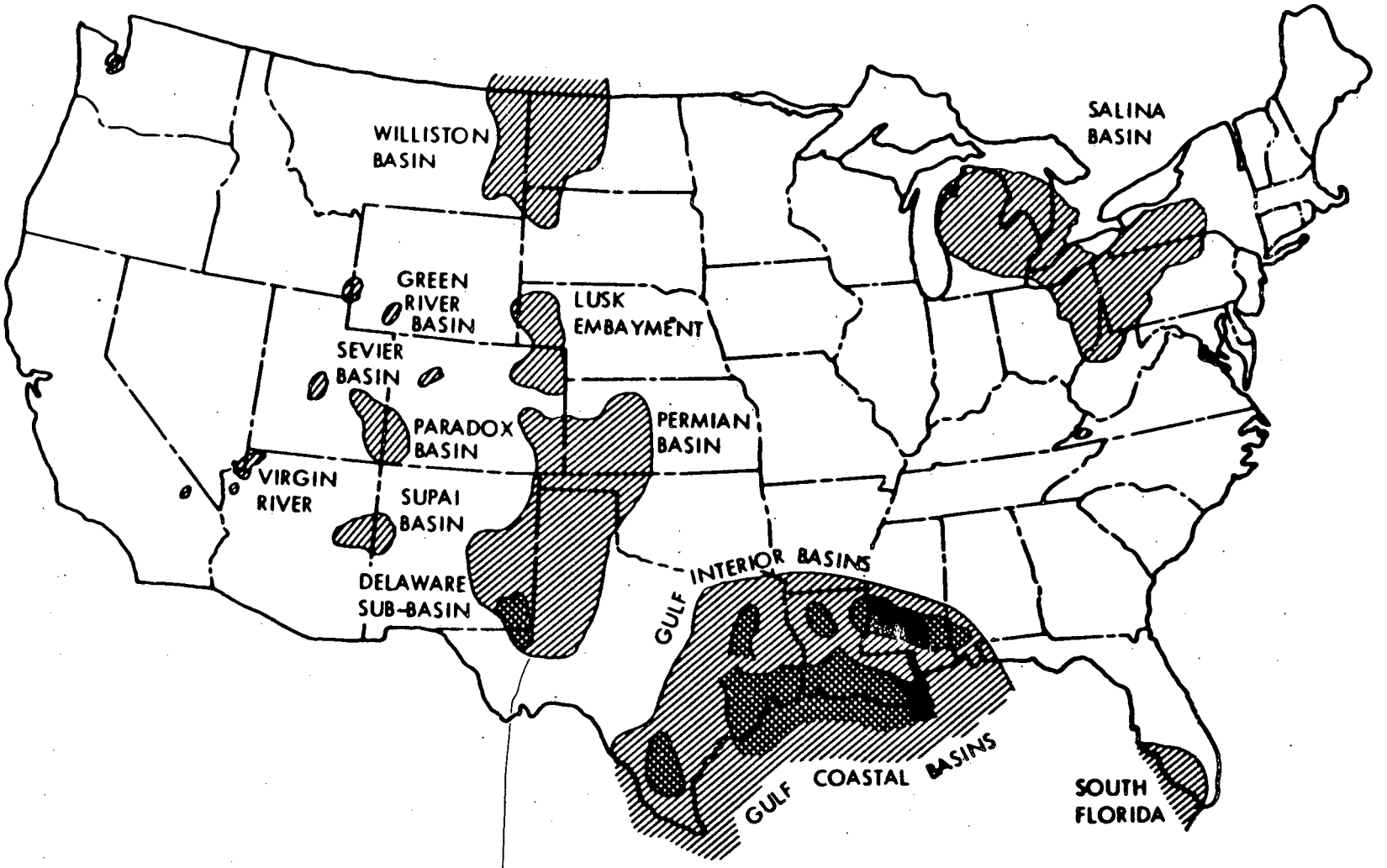



Fig. 2.6. Rock Salt Deposits in the United States. [Fig. 4 in "Proceedings of the Symposium on Waste Management, Tucson, Arizona, October 3-6, 1976," CONF-761020, R. G. Post (editor), University of Arizona, the Arizona Atomic Energy Commission and the Western Interstate Nuclear Board.]

It is assumed that the waste canisters would be shipped by rail, one cask per rail car. Upon receipt at the canistered-waste facility, the surface of the casks and the internal coolant would be checked for radioactive contamination. Assuming no such contamination existed,* the cask would be placed in the cask unloading pool. Once the cask had been submerged in the water-filled unloading pool, a special crane would be used to open the cask, remove the canister and place it on a rack which would then be transferred through a water canal to the surge pool. The surge pool would provide a means of removing these wastes from shipping casks and temporarily storing canisters until they could be packaged.

Canisters would be transferred from the surge pool to packing cells through one of two wet transfer canals. In the packaging cell, the canisters would be dried with forced air and placed in an overpack canister by use of overhead cranes. The top of the overpack canister would be welded in place. The space between the canister and its overpack would be evacuated of air and charged with helium. The overpacked canisters would be inspected for contamination and leaks, then decontaminated and sealed if necessary. The overpacked canisters would then be transferred to the holding area by motorized carts. The canister and overpack together would provide two containment barriers. In the event that the canister were breached, a second, larger overpack would be placed over the first overpack in order to maintain the two containment barriers.

The encapsulated canistered wastes would be transferred from the canistered-waste building to the subterranean storage areas through a canistered-waste shaft. At the mine level, the shaft would provide access to the TRU-ILW and hulls disposal area at one elevation and to the HLSW disposal area at a lower elevation. These two levels are assumed not to overlap. This is a conservative assumption in assessing the burial area required for waste storage. The encapsulated wastes would be transported through the shaft by a special cage. Safety features would be incorporated to prevent the cage from falling to the bottom of the shaft in the case of equipment failure.

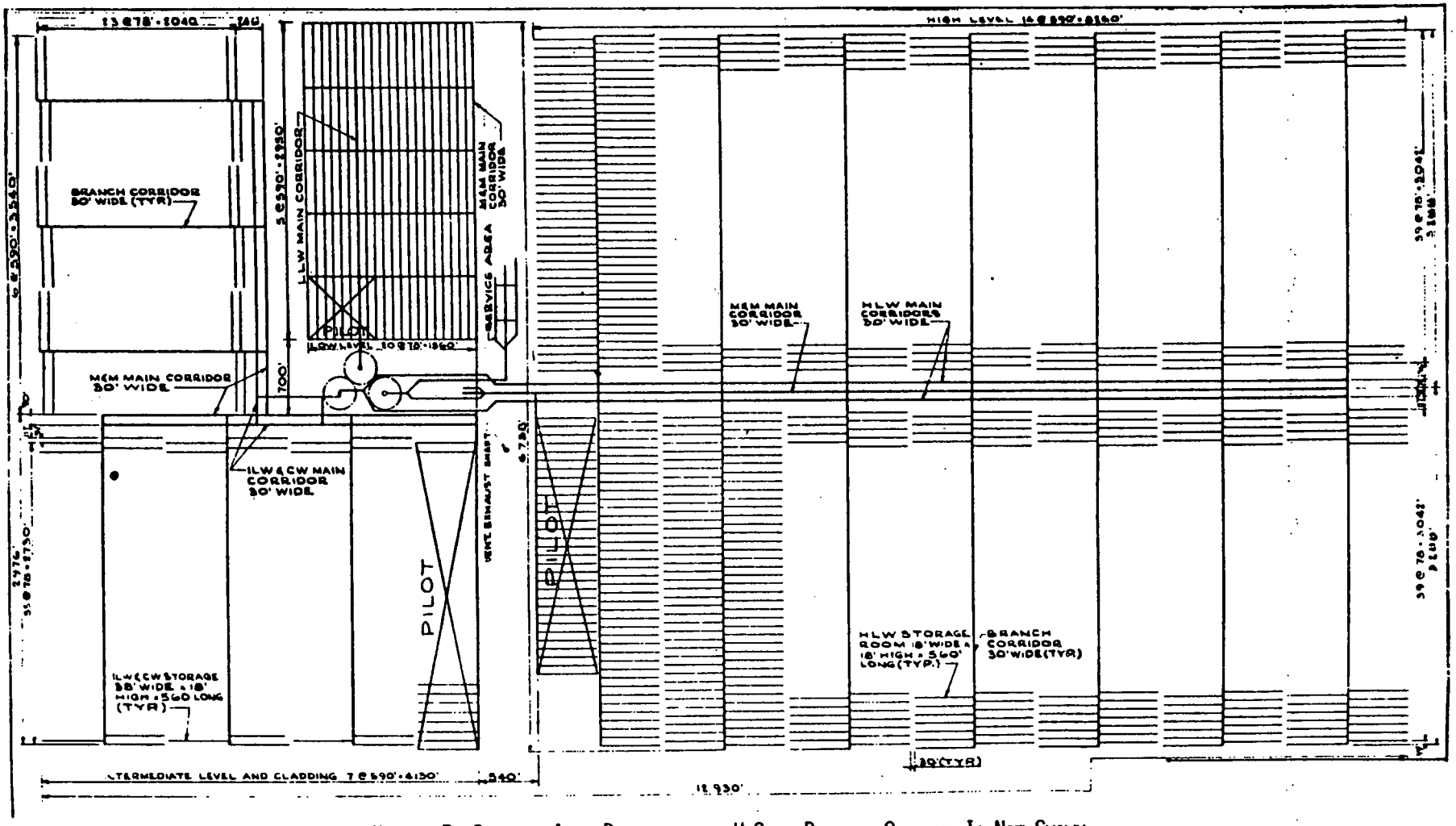
After transport through the shaft to the mine, a waste canister would pass through a mine-level receiving station before entering the storage area. The receiving station would be a shielded enclosure with a viewing gallery. A top view of the mine storage corridor and room arrangement is given in Figure 2.7. Each waste type would be placed in its own section of the mine. The spiked $PuO_2$, hulls, TRU-ILW, and TRU-LLW are assumed to be on one level, with the HLSW at a lower level.

A special shielded transporter vehicle would receive a canister at the appropriate receiving station, transport it to the proper storage room, and deposit it in a hole of appropriate size drilled in the floor. For the first few years of operation, the repository probably would be operated as a pilot facility in a retrievable mode. During this time, the storage holes would be lined with a steel sleeve and the storage rooms would not be backfilled (Fig. 2.8). This would allow for removal of the wastes in the event of abnormalities. If the repository were operating according to plan after this time, the retrievable mode would be ended--no sleeves would be used, and the storage rooms would be backfilled with mined salt within 90 days after they were filled with waste canisters.

The mine ventilation system would have to be sufficiently diverse to accommodate active excavation, disposal, and in the case of $PuO_2$, storage and possibly recovery within about 25 years. Some of the heat generated by the canistered wastes would be transferred to the mine air, which would be monitored for continuous work conditions. Exhaust fans would always maintain negative pressures in the mine relative to the atmosphere so as to ensure (1) proper ventilation of the mine and (2) proper filtration of air exiting the mine. To help maintain this negative pressure and to ensure that ventilation air flowed only in the desired direction, the entire exhaust system would be fitted with backflow preventors. After filtration through prefilters and high efficiency particulate air (HEPA) filters, the exhaust air would be discharged to the atmosphere through a stack continuously monitored to detect any radioactivity or noxious gases in the air stream. This central filter station, operating in conjunction with the mine air supply system, would provide confinement for all the mine air. In order to enhance dilution and dispersion, all ventilation air from the surface buildings containing waste or waste-handling facilities also would be exhausted through the ventilation exhaust facility serving the mining area.

The primary cooling water and air systems of the canistered-waste building would consist of closed loops designed to provide a positive barrier against potential leaks of radioactive materials to the environment and to personnel areas. The primary cooling water system would be backed up by an emergency system supplying cooling water for emergency utilities and system operation. Such an emergency heat sink would provide ample cooling in off-normal conditions.

*Casks found to have surface contamination or to contain breached waste canisters would be subjected to special handling and decontamination procedures. These procedures are not detailed in this report.




NOTE: PU STORAGE AREA REQUIRED FOR U-ONLY RECYCLE OPTIONS IS NOT SHOWN.

Fig. 2.7. Mine Master Plan for Reprocessing Wastes Repository. (From "Waste Isolation Facility Description - Bedded Salt," Office of Waste Isolation, Y/OWI/SUB-76/16506, September 1976.)

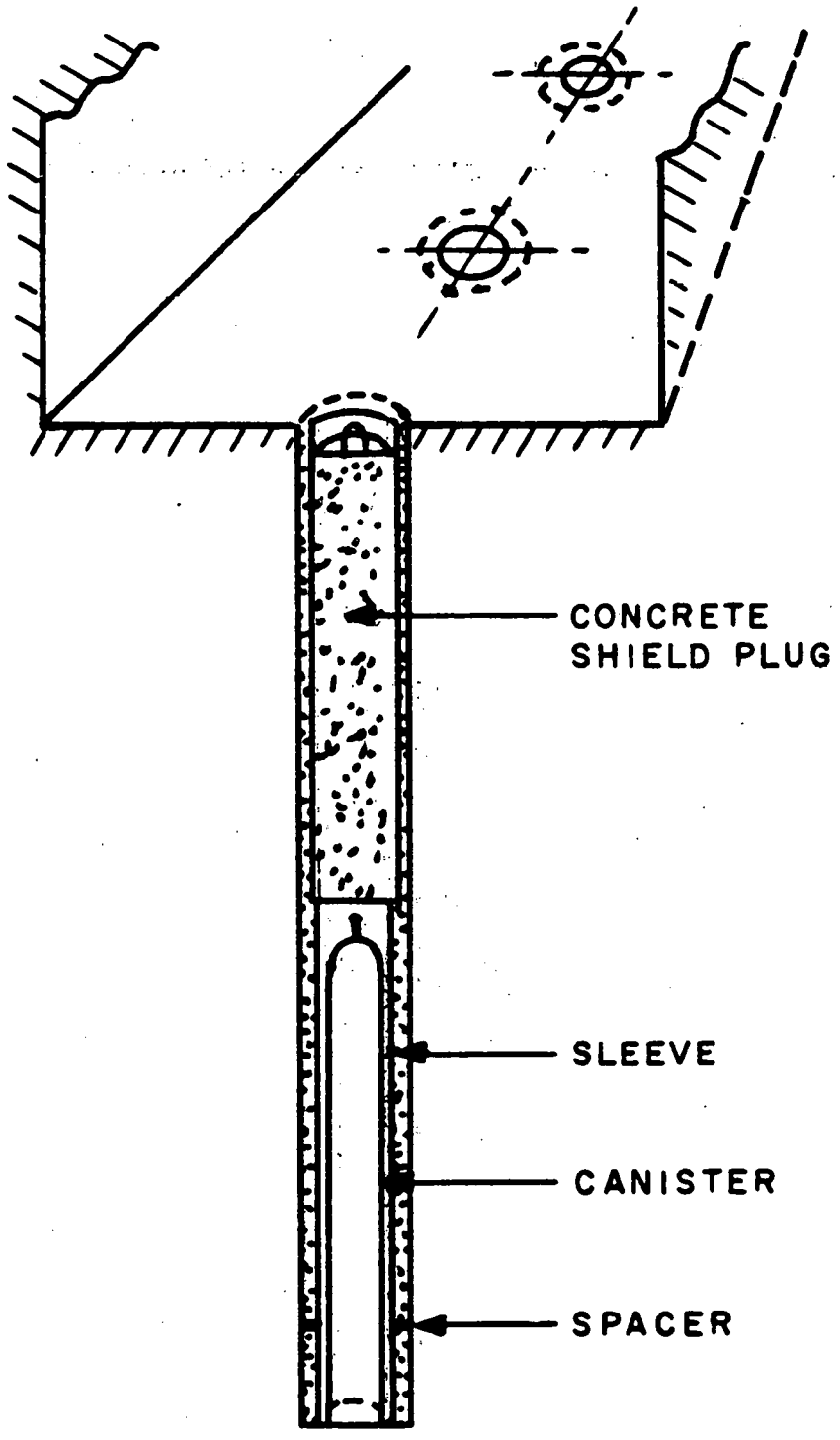



Fig. 2.8. Retrievable Emplacement. (From: "Waste Isolation Facility Description - Bedded Salt," Office of Waste Isolation, Y/OWI/SUB-76/16506, September 1976.)

The heating, ventilation, and air conditioning (HVAC) system for the facility would be designed to supply properly conditioned air to operational areas, to ensure that air was restricted to prescribed flow paths for confinement, to pass the airflow through final filters or treatment systems, and then to discharge the filtered air through a stack to the environment. The building structures and ventilation systems would provide confinement of radioactive materials and ensure that personnel exposure was maintained as low as reasonably achievable.

The operation of the surge pool would generate both liquid and solid radioactive wastes, which would have to be collected, treated, packaged, stored, and disposed. A number of auxiliary systems would be devoted solely to the handling and treatment of such wastes in an environmentally safe manner. These wastes are assumed to be low-level, nontransuranic.

The major sources of liquid and semiliquid radioactive wastes would be the water treatment system and the cask cool-down and decontamination system. The next most important source of contaminated waste would be equipment and facility decontamination and flush solutions. The liquid radioactive wastes and filter sludges would be concentrated in a waste evaporator, and the concentrates, as well as spent ion-exchange resins and contaminated air filter cartridges, would be immobilized in a solid matrix of cement or other suitable material. The waste and solidification agent would be mixed and packaged in a container, such as a 55-gallon drum, and capped. The container then would be stored onsite to await final disposal. Radioactive gases released from cask decontamination or fuel pool operations would be collected through HEPA filters, condensers, or advanced systems.

#### 2.2.1.2 TRU-LLW Facilities

It is assumed that transuranic low-level waste (TRU-LLW) would be generated by the reprocessing of spent fuel. The TRU-LLW would be loaded in 55-gallon drums without first being compacted or incinerated. Palletized loads of the containers would be shipped to the TRU-LLW receiving building by truck or rail. The carrier would enter the building and be placed on a transfer car for transport through an airlock to the unloading room, where the carrier would then be emptied. Lift trucks would transport the pallets of waste containers to the mine shaft. The TRU-LLW mine shaft would provide access for transport of the waste pallets from the surface building to the TRU-LLW subterranean receiving station, where the pallets would be loaded onto a transporter and moved to the storage area. The mine for low-level waste would be very similar to that for canistered waste. However, in the case of TRU-LLW, the waste containers would be stacked in the rooms rather than buried in holes in the floor.

#### 2.2.1.3 Plutonium Storage/Disposal Facilities

To date, there are no conceptual designs for a deep geologic storage facility for $PuO_2$. Surface storage of pure $PuO_2$ has been described for the U-only recycle cases in GESMO.[4] The exact methods and procedures for storage or disposal of the spiked $PuO_2$ assumed for this study have not been considered in detail; however, a hypothetical underground facility is outlined below.

The bulk of $PuO_2$ shipment to date has been by truck. It is expected that 61-cm (2-ft) long containers packed with 6 kg (13 pounds) of $PuO_2$ would be used. The overpacking, similar in size and shape to a 55-gallon drum, would be designed to prevent criticality in any packing geometry. It is assumed that a $PuO_2$ facility would consist of a separate receiving building, handling facility, hoist, shaft, and mine. The support buildings and systems of the main repository would be used for the $PuO_2$ facilities, with the possible exception of the ventilation system.

For the system conceptualized for this report, each 6-kg (13-pound) canister of spiked $PuO_2$ would be stored in a metal-lined underground cavity (much like the arrangement used for canistered waste) while still overpacked. The geometry of corridors and storage cavities would be similar to the canistered-waste mines, except the distance between cavities would have to be different to accommodate the different heat loading. After being loaded with a unit of spiked $PuO_2$, a cavity would be temporarily sealed airtight. If retrieval of the $PuO_2$ was later desired, the air in the cavity would be tested to ensure that the inner canister and overpack had not failed. If both had failed, the unit would remain in the cavity and await backfilling while other units were removed. If the barriers had not failed, the unit would be transported to the handling facilities for shipment.

If recycle of $PuO_2$ did not occur within a suitable period (depending on the predicted integrity of the units), then the mine would be backfilled and the facilities decommissioned. Assuming recycle did occur, two options would be available for mine ventilation, the choice of which would affect recovery. If ventilation of a corridor ceased after that corridor was full, a cool-down time might be necessary before recovery procedures could begin. Such a cool-down period would not be necessary if ventilation of a corridor continued after the corridor was full.

The storage or disposal of $PuO_2$ in a geologic medium would present special problems of criticality. For criticality to occur, the canisters and overpacks bearing the spiked $PuO_2$ would have to be leached, with enough $PuO_2$ leaking out of the canisters and coming together to form a critical mass. Even though this event is highly improbable, analyses have been done to find the minimum thickness of a slab and the minimum radius of a sphere to achieve criticality for various $PuO_2$ solutions in salt. Graphs of these analyses are shown in Figures 2.9 and 2.10. These figures are for pure $PuO_2$ in salt solutions and do not include the 5% fission products in the spiked $PuO_2$ mixture. These calculations are applicable only in the absence of neutron-absorbing fission products. Even a small amount of fission products would increase the mass requirement for criticality. Thus, Figures 2.9 and 2.10 should be viewed as being conservative calculations for storage of the $PuO_2$ in the manner assumed in this report. Events which would cause the $PuO_2$ to come together in the amounts necessary for criticality are highly unlikely, even over the long time period involved.[6]

### 2.2.2 Spent Fuel Storage/Disposal

The spent fuel assemblies are assumed to be either stored near the surface of the ground or buried in deep geologic salt formations. For either type of storage, the spent fuel would be handled as shown in Figure 2.11.

The facilities and processes in the receiving building at the repository would be similar to those described for the repository for reprocessing wastes in Section 2.2.1. The spent fuel assemblies would be subject to the same general handling procedures previously described, except for the surface storage option. In that option, the assemblies would be stored near the earth's surface rather than in deep geologic formations.

#### 2.2.2.1 Surface Storage of Spent Fuel

Dry caisson storage is considered as the model interim surface storage method for packaged spent fuel. The dry caisson design adapted for this report is illustrated in Figure 2.12. This concept is under study by the Atlantic Richfield Company.[7] One fuel assembly (PWR or BWR) would be sealed in a steel canister with a 40-cm (15-inch) diameter. The packaged fuel would be filled with an inert gas (such as helium) to prevent oxidation of the canister, to promote increased heat transfer, and to provide a method of detecting leaks. This temporary storage mode could permit interim storage (up to 25 years) while a decision was being made on whether to treat the spent fuel as a resource for reprocessing or as a waste requiring permanent disposal.

There would be three confinement barriers for this method of storage: the fuel cladding, the fuel canister, and the hole liner and shield plug. The hole liner and shield plug would provide protection against entry of water. The hole liner would consist of corrosion-resistant materials, such as concrete. Caisson storage would utilize the earth for passive cooling and shielding by placing nuclear material into lined holes in the earth's surface. The decay heat transferred to the earth would eventually be conducted to the earth's surface and then dissipated to the atmosphere.

The canister would be stored inside a carbon-steel well casing, or caisson, which might range from 50 to 100 cm (20 to 40 inches) in diameter. Larger diameters might be used to reduce the heat flux into the earth. To provide adequate shielding, the caisson would extend about 7.6 m (25 ft) into the ground and would be fitted with a high-density metal or concrete shielding plug. Caisson covers could be sealed by any of several methods to provide protection against unauthorized removal. The caissons are assumed to be placed 7.6 m (25 ft) apart in a square array. A security fence would surround the storage area.

The thermal characteristics of the geologic features of the surface interim storage site would affect the capacity of the caisson to dissipate heat. Caissons probably would be located in areas where the water table was substantially lower than the caisson. In addition, the area should not be susceptible to flooding, seismic, tornado, or sabotage events. Isolated arid regions would probably be well suited for caisson storage yards.

The final design of dry storage facilities would be subject to siting and licensing procedures. Design standards would have to accommodate efficient and economical plant operation. However, the facility might contain in excess of $10^9$ curies of fission products, so the design of systems, structures, and components also would have to account for the possibility of uncontrolled releases of radionuclides. In general, the safe storage of irradiated fuel depends on the integrity of the fuel cladding as the primary barrier to the release of radionuclides.

For this report, it is assumed that the surface area of a surface-storage spent fuel repository would be the same as for an underground reprocessing wastes repository. That is, the spent fuel storage area would be 800 ha (2000 acres) and would be surrounded by a buffer zone of an

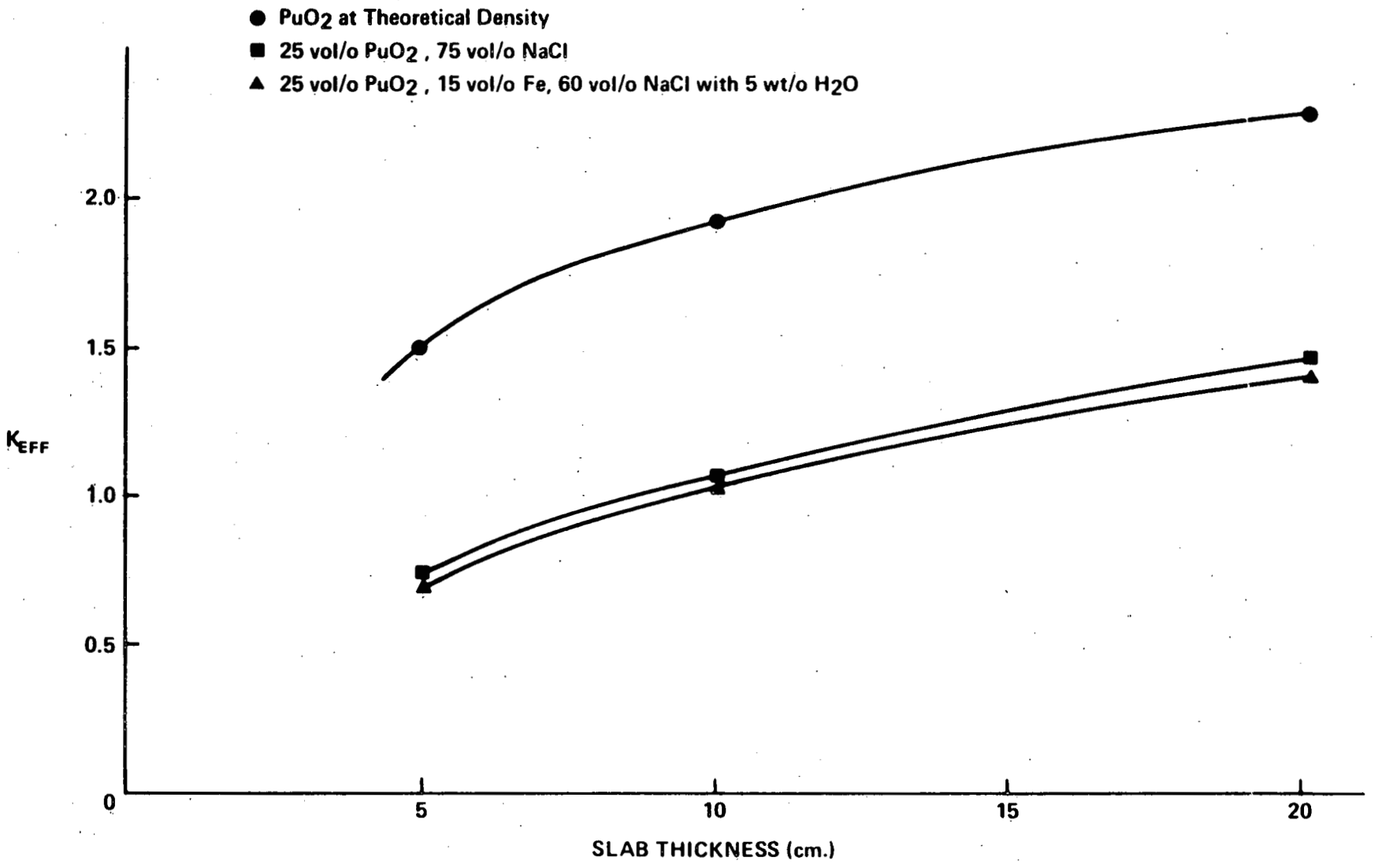



Fig. 2.9. $K_{eff}$ for $PuO_2$ in a Salt Repository, Slab Geometry. [Fig. 4.12 in "Public Comments and Task Force Responses Regarding the Environmental Survey of the Reprocessing and Waste Management Portions of the LWR Fuel Cycle (NUREG-0116)," U.S. Nuclear Regulatory Commission, NUREG-0216, March 1977.]

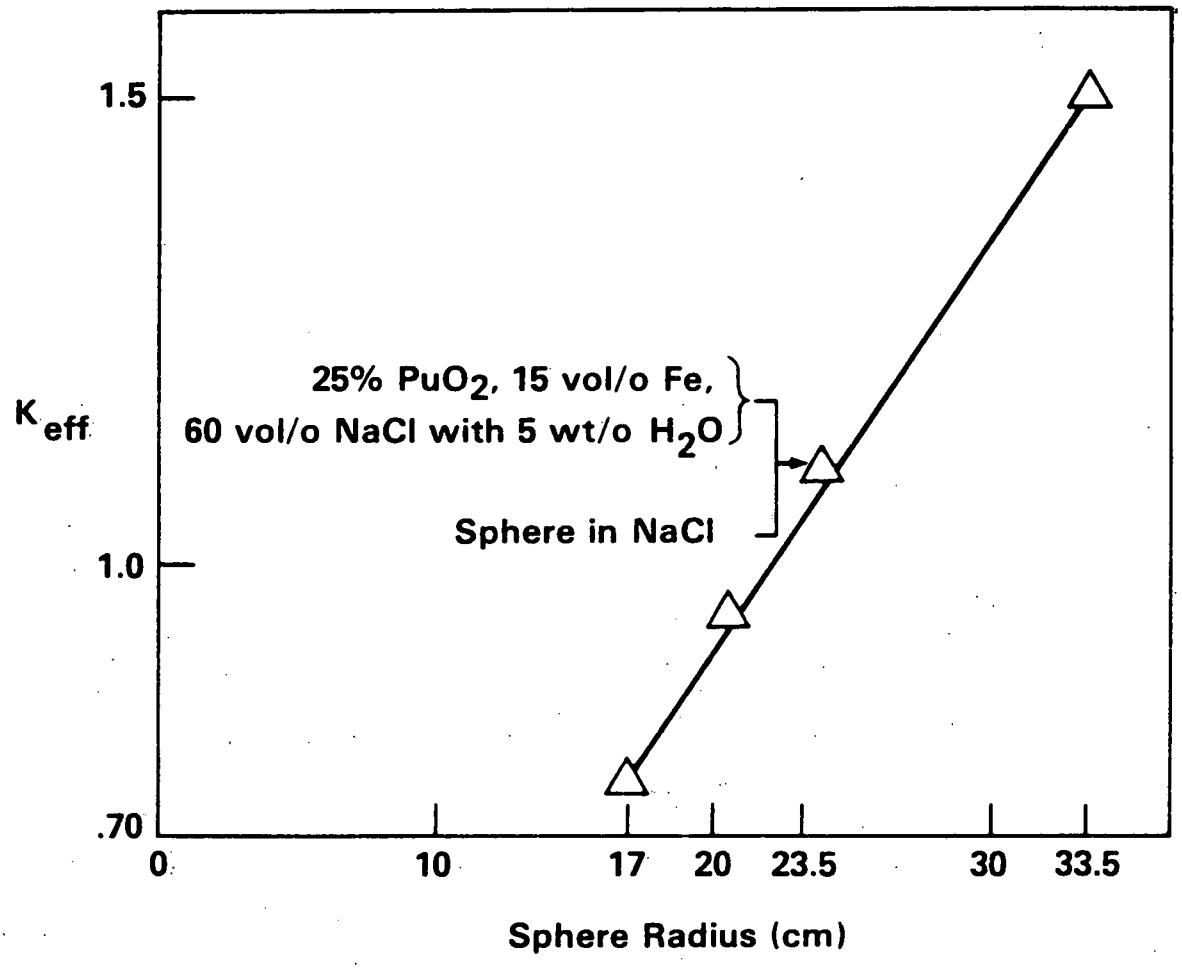



Fig. 2.10. $K_{eff}$ for $PuO_2$ in a Salt Repository, Sphere Geometry. (Fig. 4.13 in Environmental Survey of the Reprocessing and Waste Management Portions of the LWR Fuel Cycle," U.S. Nuclear Regulatory Commission, NUREG-0116, October 1976.)

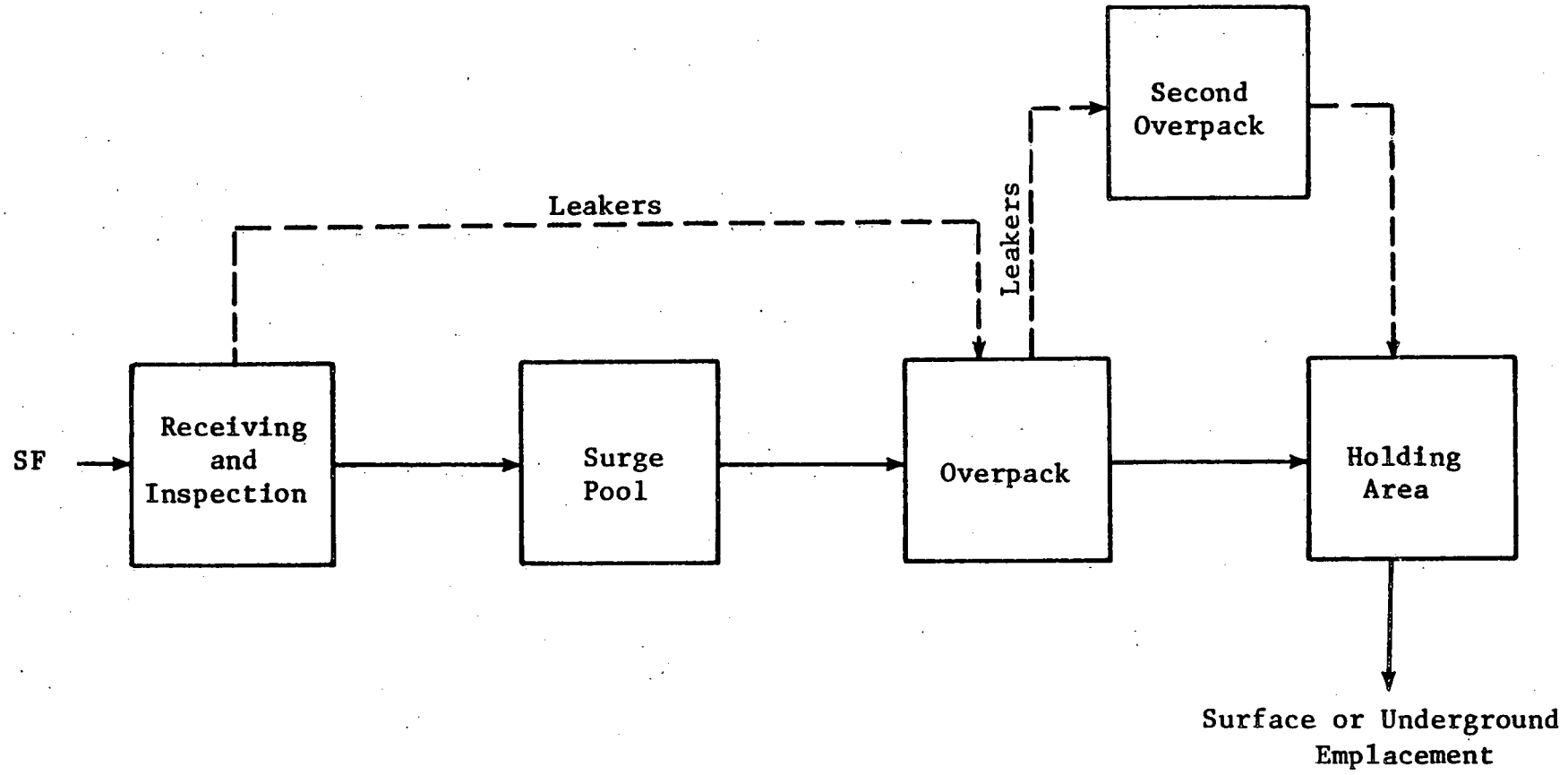



Fig. 2.11. Flow Diagram for Spent Fuel (SF) Repository.

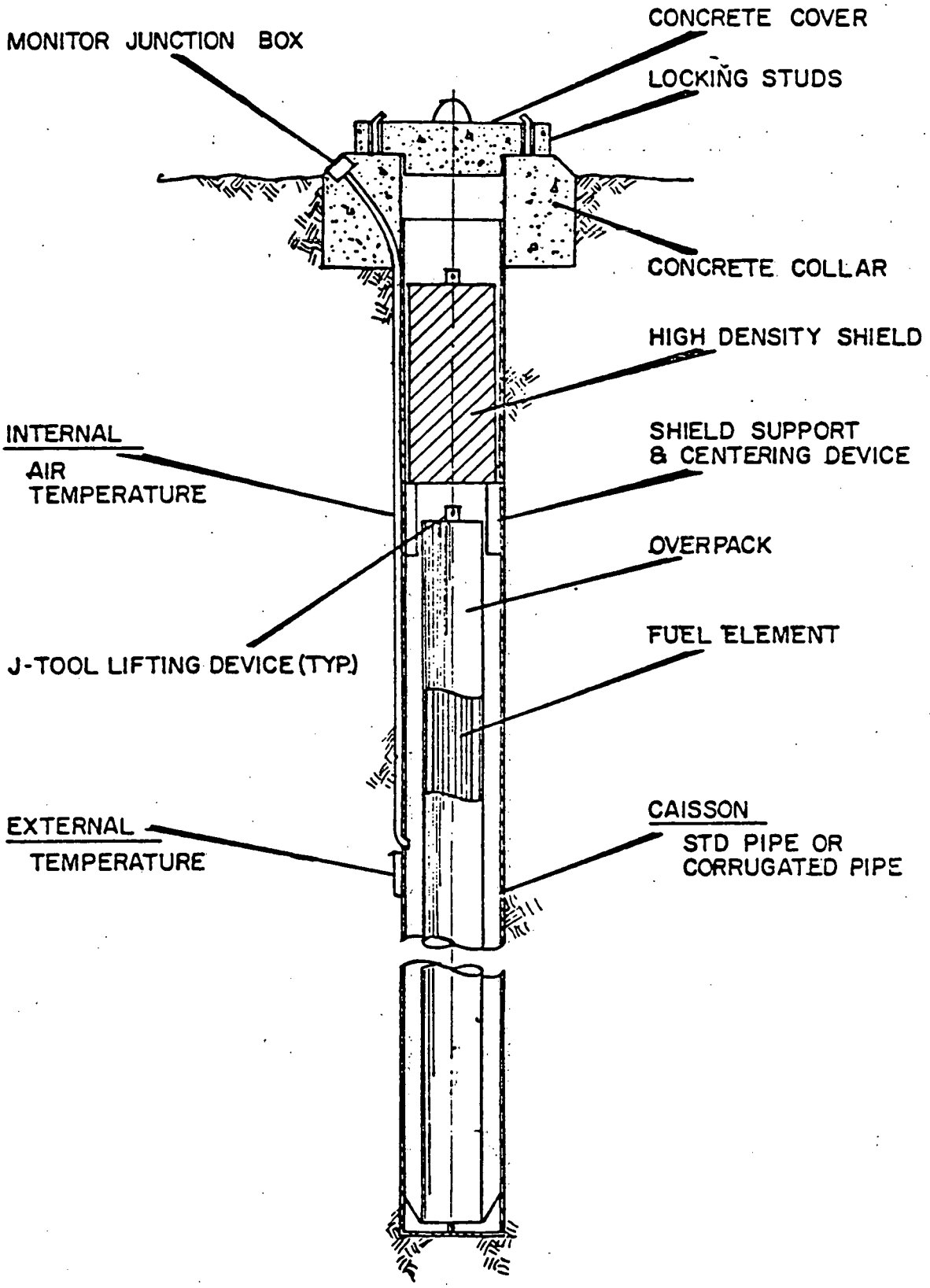



Fig. 2.12. Spent Fuel Storage Caisson.

additional 1200 ha (3000 acres). Hence, the total areal allotment to a dry, surface-storage repository would be 2000 ha (5000 acres).

Heat dissipation rates for caisson storage would be a function of the time the spent fuel had been in pool storage, the thermal conductivity of the soil at the site, and the diameter of the caisson. Preliminary analysis indicates that the ground could safety dissipate approximately 1.5 kilowatts thermal per caisson without exceeding 370°C (700°F) at the cladding. This heat level is equivalent to the decay heat generated by a PWR assembly four years out of the core or a BWR assembly two years out of the core.[8] A preliminary analysis of the heat distribution for caisson storage recently performed by the Atlantic Richfield Hanford Company may provide approximations of the heat distribution near the caissons.[8] The results are based upon Hanford soil temperature data.

Maximum canister temperatures are shown in Figure 2.13 as a function of spacing for PWR fuel at three, five, and ten years after discharge. For ten-year-old spent fuel at a canister spacing of about 5.5 m (18 ft), the maximum canister temperature would be about 175°C (350°F). This corresponds to a maximum cladding temperature of about 245°C (475°F). Since the canister spacing in the model caisson storage facility is assumed to be 7.6 m (25 ft), a 5.5-m spacing calculation will conservatively account for the area needed for roadways, equipment, etc. The isotherms resulting from ten-year-old fuel spaced at 5.5 m is illustrated in Figure 2.14. Based on an 18°C (64°F) ambient soil temperature, the rise in surface temperature as a result of ten-year-old PWR fuel stored in a caisson would be less than 12°C (20°F). The temperature rise would be even less for ten-year-old BWR fuel.

### 2.2.2.2 Deep Geological Storage of Spent Fuel*

Spent fuel can be stored in deep geological repositories, either in retrievable (stowaway) or nonretrievable (throwaway) modes. The retrievable mode would permit removal of the spent fuel in the future for reprocessing. For retrievable storage, the storage rooms would not be backfilled, and liners would be inserted in the holes drilled in the floor of the storage rooms. For the nonretrievable disposal of spent fuel, there would be no liners in the holes and the rooms would be backfilled. The geological storage facility for both options would be similar to that for reprocessing wastes.

The deep storage facility would contain separate storage rooms for BWR and PWR spent fuel, haulageways, access shafts, ventilation tunnels and shafts, service areas, and temporary holding facilities. There could also be a storage area for nontransuranic low-level waste; however, such a storage area is not included in the model facility for this report.

The mine layout and storage sequence for the spent fuel storage facility would be of a conventional room-and-pillar design incorporating the requirements of mine ventilation, mine opening stability, heat dispersal, and efficient use of mining and transport equipment. The facility would consist of a dendritic pattern of waste storage rooms and corridors surrounding a set of five shafts (Fig. 2.15). All wastes would be lowered through a special shaft connected to haulageways for transport of waste to either the BWR or the PWR storage areas. The spent fuel canisters would be transported from the spent fuel building through the shaft to the emplacement holes in a manner very similar to that described earlier for the canistered wastes at the reprocessing wastes repository.

For the first few years of repository operation in the throwaway option, all wastes would be retrievable. During this time the storage holes would be lined with a steel sleeve and the storage rooms would not be backfilled. This would allow for removal of the spent fuel in the event of abnormalities. If the repository was operating according to plan after this time, the retrievable mode would be ended and no sleeves would be used. In addition, the storage rooms would be backfilled with mined salt within 90 days after they were filled.

Engineering precautions would have to be taken if retrievability was to be maintained for at least 25 years (stowaway option). In this option, the spent fuel assemblies would be emplaced in a manner similar to that used for the throwaway option. That is, the storage holes would be lined with steel and the storage rooms would not be backfilled. The steel liners would at least temporarily protect the canisters against the corrosive salt environment, and the cylindrical shape and the freespace around the liners would provide protection against the squeezing action expected to be exerted by the heated salt.** Rooms filled with retrievable wastes would be

*Much of the information used in this section was obtained from Reference 9.

**These problems could be of such magnitude that if long-term retrievability were to be an option, a repository constructed in igneous rock might be preferred.

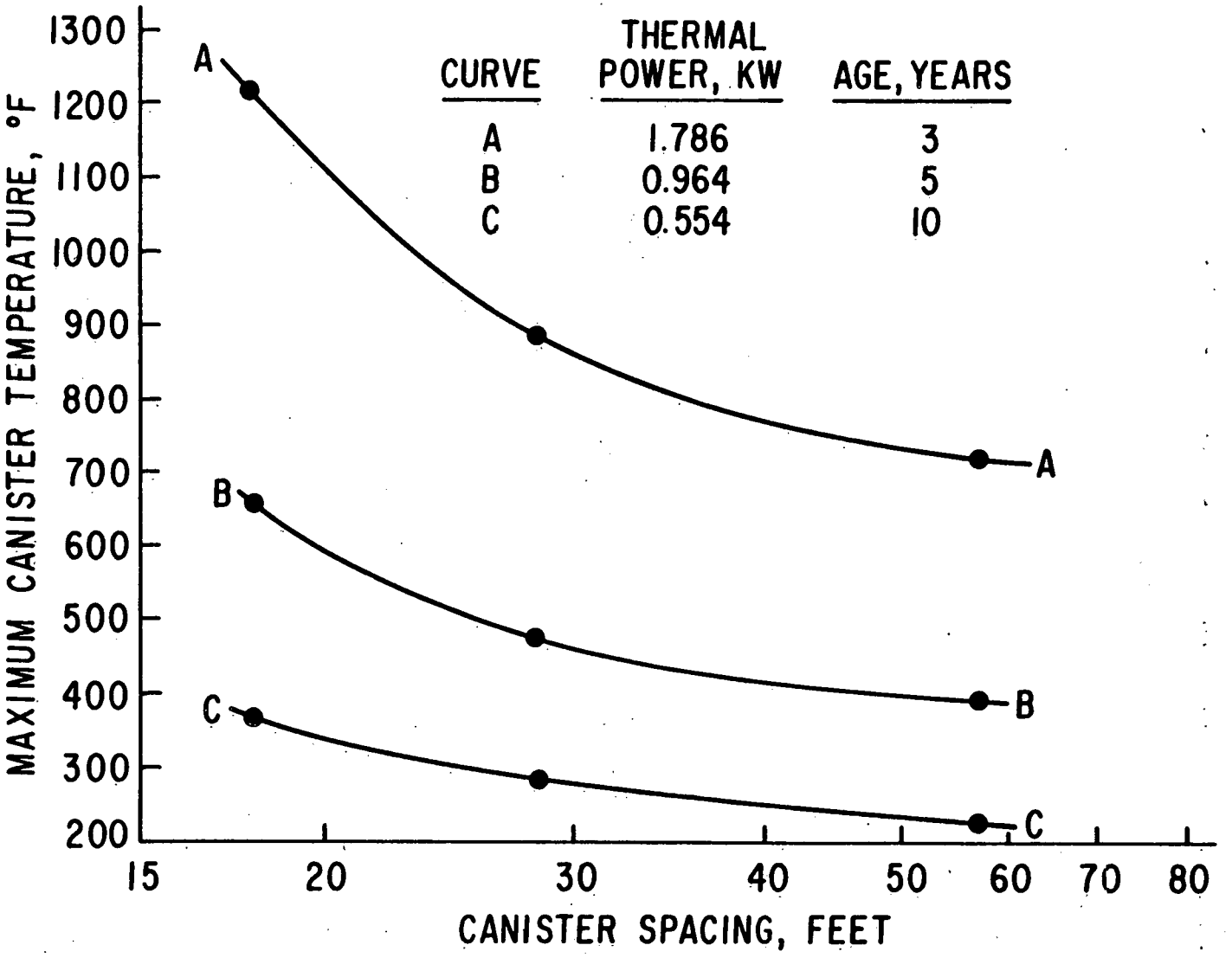



Fig. 2.13. Canister Temperature vs. Spacing for PWR Fuel (caisson storage). [From "Spent Unreprocessed Fuel Facility, Engineering Study of Storage Concepts (Draft)," prepared by Atlantic Richfield Hanford Company for U.S. Energy Research and Development Administration, 1 June 1977.]

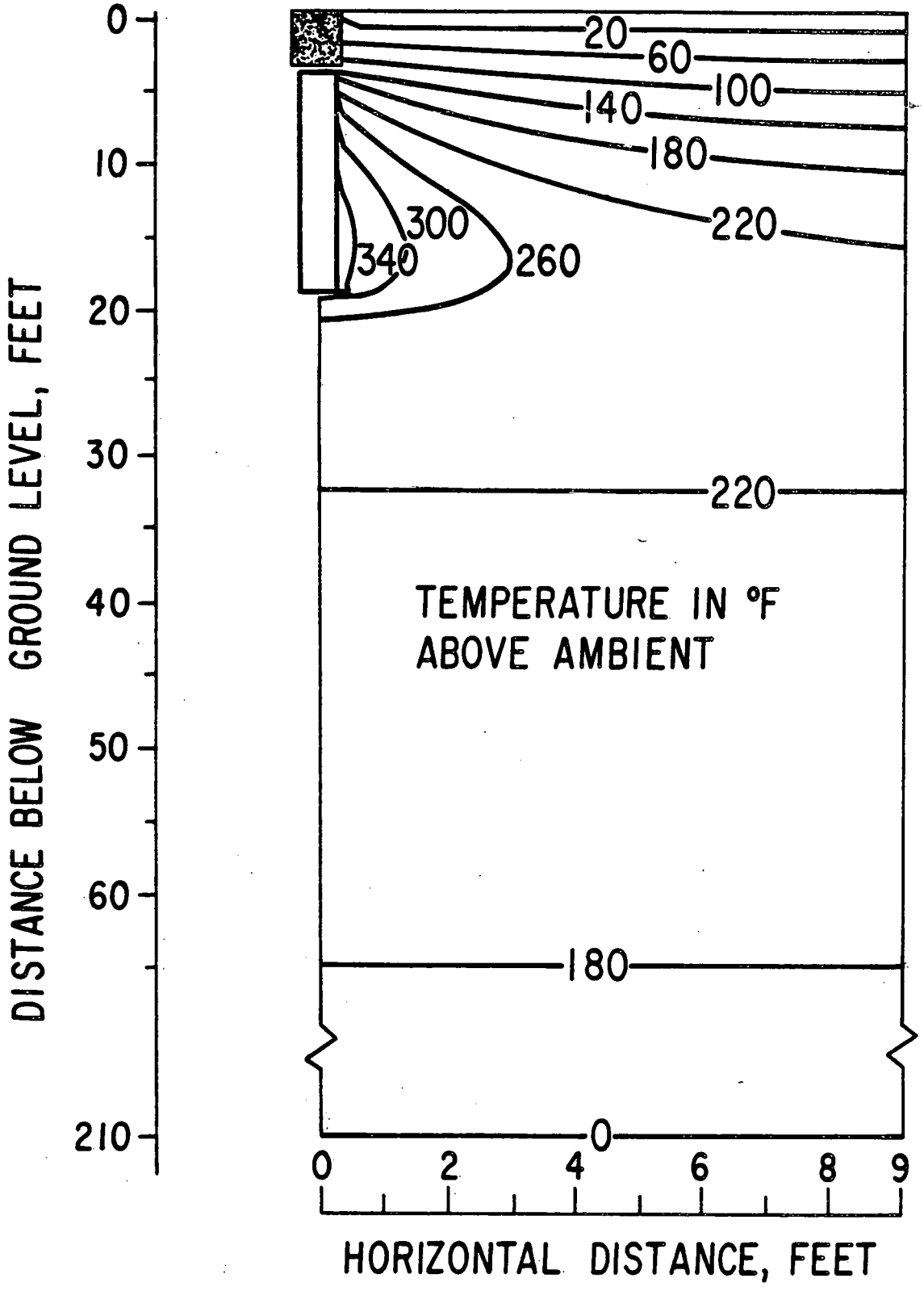



Fig. 2.14. Caisson Isotherms for PWR Fuel Ten Years after Discharge (spacing of 5.5 m). [Adapted from "Spent Unreprocessed Fuel Facility, Engineering Study of Storage Concepts (Draft)," prepared by Atlantic Richfield Hanford Company for U.S. Energy Research and Development Agency, 1 June 1977.]

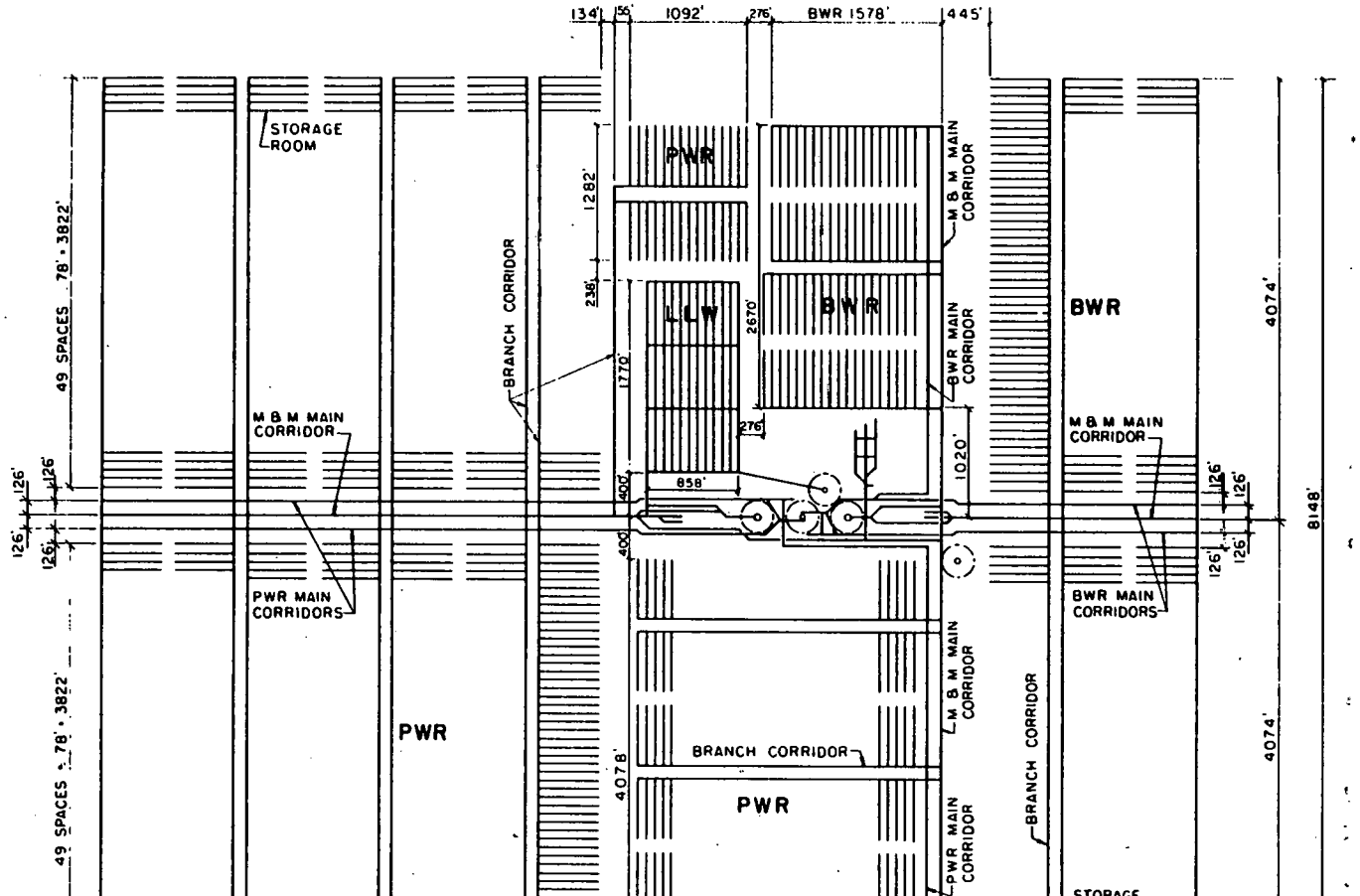



Fig. 2.15. Mine Master Plan for Spent Fuel Repository. (From "Waste Isolation Facility Description for the Spent Fuel Cycle," Office of Waste Isolation, Y/OWI/SUB-77/16543, May 1977.)

sealed from the remainder of the mine. Except for inspection or retrieval of waste, no filled storage room would be ventilated.

The retrieval procedure could be accomplished in two steps. First, the end of the storage room would be opened. The air then drawn into the rooms would cool the room and dilute the radioactive gases which might have accumulated. The time necessary for cooling the room to the level that men and machinery would be able to retrieve canisters would be primarily a function of the canister power and gross heat loadings.[10] The second step, actual retrieval of the waste canisters, would simply be a reversal of the emplacement procedure.

The ventilation supply for the mine would be provided through ventilation supply ducts in the shafts. The supply airflow rate would be monitored and an alarm would be activated if flow fell below an allowable minimum. The entire facility would be operated at a negative air pressure relative to atmospheric by adjustment of the supply fan pressures relative to the exhaust pressures. Exhaust air would be filtered and vented to the surface.

General corrosion in the salt mine environment could be a potential problem. If corrosion did occur, it is likely that first canisters, then fuel claddings would fail in a random combination. For regular occurrences of this nature, it would be necessary to treat the mine ventilation air to remove the airborne radionuclides when the retrievable storage areas were purged for the purpose of reclaiming the spent fuel after canister corrosion. Careful ventilation system design, judicious decisions on the order of repository rooms to be filled, and prompt backfilling would minimize any contamination of the mine from radioactive gases. If monitoring indicated that a problem was developing, temporary airtight seals could be placed at the junction of the branch corridors with the main corridors.

Because of the presence of fissile elements (uranium and plutonium) in spent fuel assemblies, precautions would have to be taken to avoid a criticality incident. The handling of spent fuel assemblies prior to emplacement should be done in a safe and expedient manner. Much experience does exist in the handling and storage of spent fuel assemblies. Designs incorporating such features as neutron-absorbing racks, separation between spent fuel assemblies, and limitations in neutron moderation should make the chances of a criticality incident remote.

For criticality to occur after emplacement, it would be necessary for the fissile elements to migrate towards a central location. The mass requirement for criticality would depend upon the specific isotopes involved, the presence of neutron-absorbing fission products, the presence of water for moderation, and the characteristics of the repository medium. The concentrating of fissile elements in the repository could result from either a catastrophic event (earthquake) or from a large influx of water. Repository site selection should minimize the potential for such events. In any case, it would be necessary for the canister and fuel cladding both to fail before the fissile elements could migrate. Even with the failure of the canister and cladding, the possibility of criticality would be remote.

### 2.2.3 Experience

There has been recent experience in emplacement of nuclear wastes in deep geologic media, both in the United States and West Germany. The most extensive work done in the United States was Project Salt Vault near Lyons, Kansas.[11] Irradiated fuel assemblies from the Engineering Test Reactor in Idaho were placed in holes in the floor of an abandoned salt mine. Over a period of 19 months, spent fuel assemblies were shipped, transferred, stored, monitored, and eventually removed. After removal, the assemblies were returned to the Idaho Chemical Processing Plant. This demonstration was conducted between 1963 and 1968.

The purpose of Project Salt Vault was to demonstrate the technology to safely handle and store spent fuel assemblies and also to examine the effects on the salt from the high radiation field. The project was successful in both regards. The canister-handling equipment was operated safely without any major difficulties. Also, considerable data were collected on the effects of emplacing solid, highly radioactive sources in a salt-mine environment. A summary of the experiments conducted and of the results is given in Reference 11.

West Germany has accrued extensive experience in disposal of radioactive waste through its Asse salt-mine project. Containers of solid low-level waste have been stored in this mine since disposal operations began in 1967, and drums of intermediate-level waste have been stored since 1972. A proposal has been made to store a limited number of burned carbide fuel elements from the AVR pebble-bed test reactor at Julich in the Asse salt mine. A solidified high-level waste test disposal is expected in the near future.[12]

Surface storage of spent fuel and radioactive wastes has been used in the United States and Canada. The CANDU* concept developed in Canada is illustrated in Figure 2.16. This mode is used to store spent fuel at Chalk River, Ontario.[13]

Fuel from Peach Bottom 1 (a prototype high-temperature, gas-cooled reactor) is stored in a subsurface vault or caisson structure at the Idaho Chemical Processing Plant in the Idaho National Engineering Laboratory (INEL). The Peach Bottom 1 fuel consists of thorium carbide and uranium carbide in a graphite matrix. This fuel must be kept dry because the carbide will react if exposed to water.[14] A diagram of a storage hole and container is shown in Figure 2.17. After a safety analysis of the Peach Bottom storage procedure, it was concluded that the dry sealed vault and fuel canisters (composed of an aluminum alloy outer wall and a steel liner) provide more than adequate fuel containment for long-term storage.[14]

The storage hole/caisson concept has also been used for high-level radioactive wastes at the Argonne National Laboratory Radioactive Scrap and Waste Facility at INEL.[15] The waste material consists principally of metal from fuel-handling and refabrication operations. The facility was first used in 1965, and through 1974 had received waste containing about 10 million curies of radioactivity. The waste is remotely loaded into a steel waste can which is then sealed and placed in a top-loading, bottom-unloading, shielded waste-handling case for placement in a waste hole by a special transporter. The storage containers can be retrieved. A detailed examination of an underground tube and container after 5½ years of use indicated that the integrity of the container was well preserved.[14]

*The CANDU reactor is a heavy water, natural uranium reactor developed in Canada.

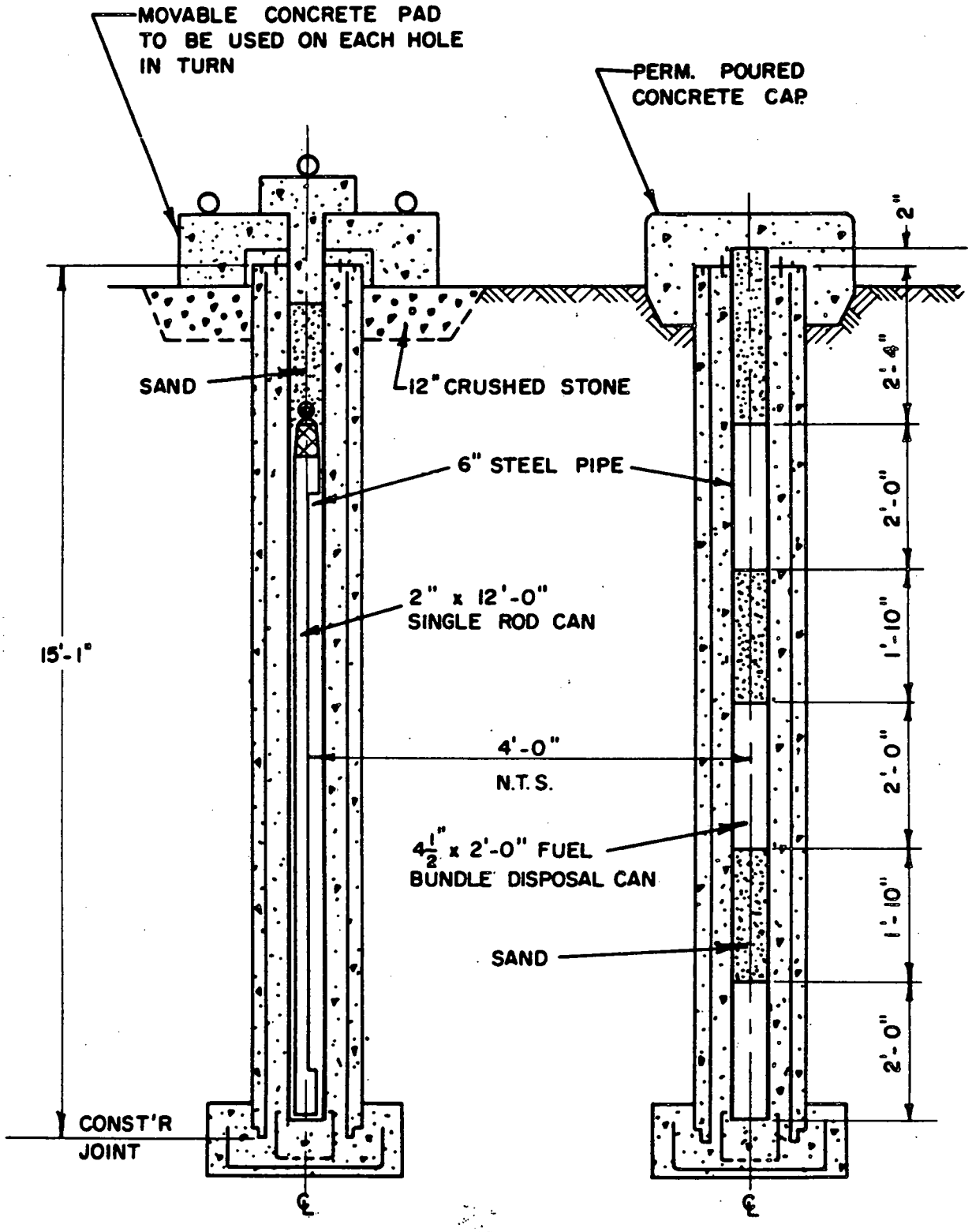



Fig. 2.16. Typical Concrete Tile Hole for the Surface Storage of CANDU Fuel. (Fig. 6 in J. A. Morrison, "AECL Experience in Managing Radioactive Wastes from Canadian Nuclear Reactors," Atomic Energy of Canada Limited, Chalk River Nuclear Laboratories, AECL-4707, March 1974.)

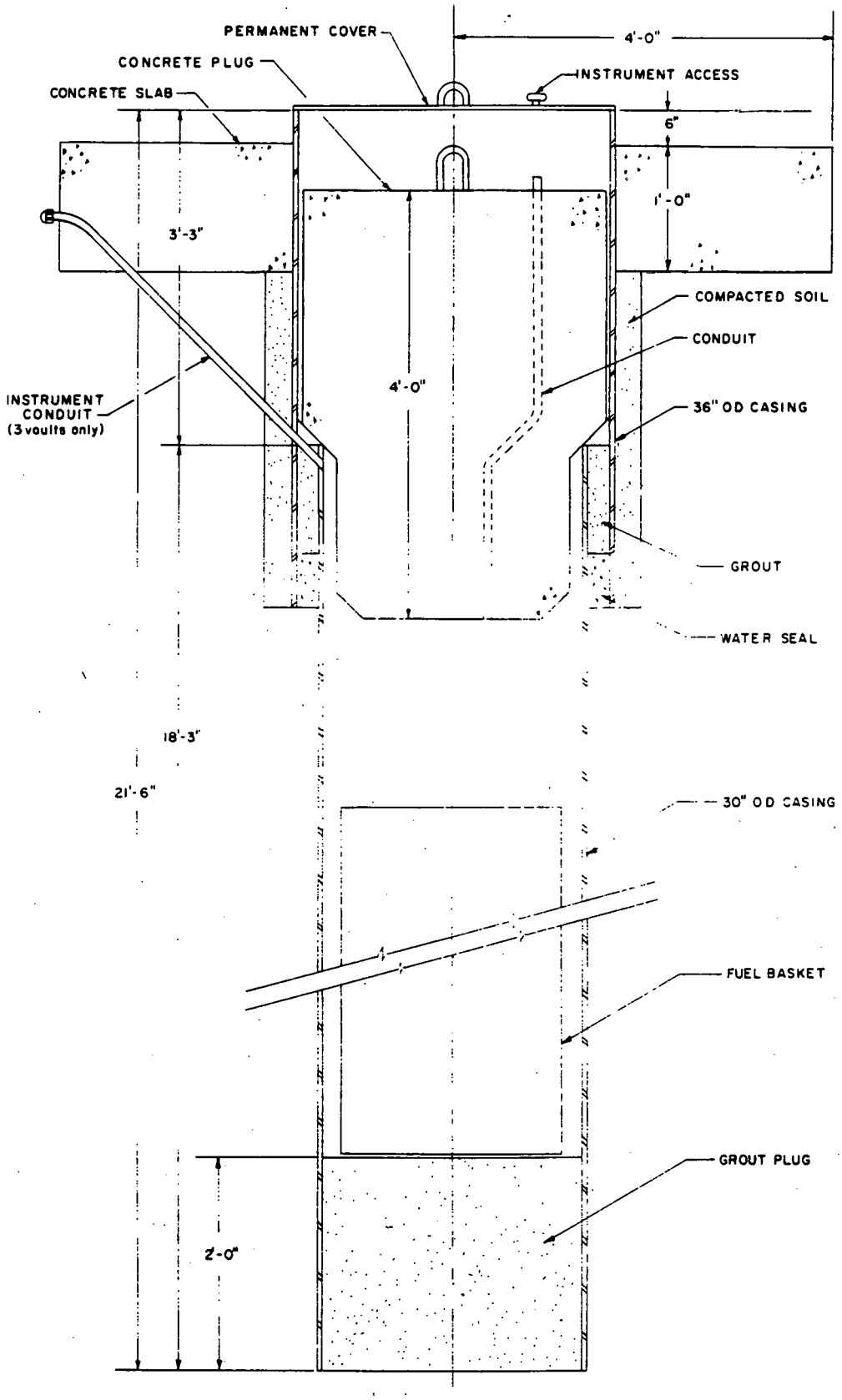



Fig. 2.17. Peach Bottom 1 Spent Fuel Storage Vault. (From J. D. Hammond, R. S. P'Pool and R. D. Morrow, "Safety Analysis Report for Peach Bottom 1 Core 1 Fuel Storage Facility," Idaho Nuclear Corporation for U.S. Atomic Energy Commission, June 1971.)

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12. K. Kuhn and J. Hamstra, "Geologic Isolation of Radioactive Wastes in the Federal Republic of Germany and the Respective Program in the Netherlands," presented at the International Symposium on the Management of Wastes from the LWR Fuel Cycle, Denver, Colorado, 11-16 July 1976, CONF-76-0701.

13. J. A. Morrison, "AECL Experience in Managing Radioactive Wastes from Canadian Nuclear Reactors," Chalk River Nuclear Laboratories, AECL-4707, Chalk River, Ontario, March 1974.

14. J. D. Hammond, R. J. P'Pool and R. D. Modrow, "Safety Analysis Report for Peach Bottom 1 Core 1 Fuel Storage Facility," Idaho Nuclear Corp., IN-1465, 1971.

15. "Draft Environmental Statement, Waste Management Operations at Idaho National Engineering Laboratory, Idaho," U.S. Energy Research and Development Administration, ERDA-1536, 1976.

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"Environmental Survey of the Reprocessing and Waste Management Portions of the LWR Fuel Cycle," U.S. Nuclear Regulatory Commission, NUREG-0116, October 1976.

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## 3. THERMAL ANALYSIS

For five of the six options considered in this report, wastes would be placed in deep geologic salt repositories. The actual number of such repositories needed would depend on the amount of each type of waste and the spacing (or density) of the waste canisters in the storage areas. Because of the heat produced by radioactive decay, the emplacement density of five types of wastes would be determined by thermal criteria.* The discussion of these criteria in this section is summarized from Reference 1.

Waste emplacement density is estimated by considering: (1) a reference site with an assumed stratigraphy and set of thermal properties, (2) five types of wastes, and (3) a comparative criterion involving maximum thermal energy. Relative emplacement densities are then calculated using this comparative criterion for various waste types.

The reference site assumed involves five unbounded horizontal layers.** The disposal horizon is assumed to be in the middle of a bedded salt layer 50 meters (160 ft) thick (see Fig. 3.1). The all-salt layer is assumed to be bounded above and below by 250-meter (820-ft) layers consisting of salt (50%) interbedded with anhydrite, shale, and dolomite. The 300-meter (980-ft) top layer and the 3150-meter (10,300-ft) bottom layer are assumed to be mixtures of limestone, sandstone, and shale. The assumed thermal properties of the reference site are given in Table 3.1. These properties correspond to the "reference case" in Reference 1.

Table 3.1. Thermal Properties of Reference Site[a]

| Layer | Density, $kg/m^3$ | Specific Heat, J/kg-°C | Thermal Conductivity, W/m-°C |
|---|---|---|---|
| Layer 3 (all salt) | 2100 | 900 | 4.53 |
| Layers 2 & 4 (1/2 salt, 1/2 other) | 2300 | 900 | 2.58 |
| Layers 1 & 5 (no salt) | 2500 | 900 | 1.81 |

[a]Bases of assumptions made in this table are:

- Approximate densities of salt and "others" are from References 8 and 9.
- Salt conductivity is from Reference 10.
- Conductivity of "others" (layers 1 & 5) is assumed to be 40% of salt conductivity (based on Reference 8).
- Conductivity of layers 2 & 4 is estimated by homogenizing conductivities of salt and "others" using the methods of Reference 10.
- Specific heat assumptions are from Reference 10.

*For other wastes, emplacement densities would be determined by mechanical criteria.
**Stratigraphic assumptions in this section are based on information provided in References 2-7.

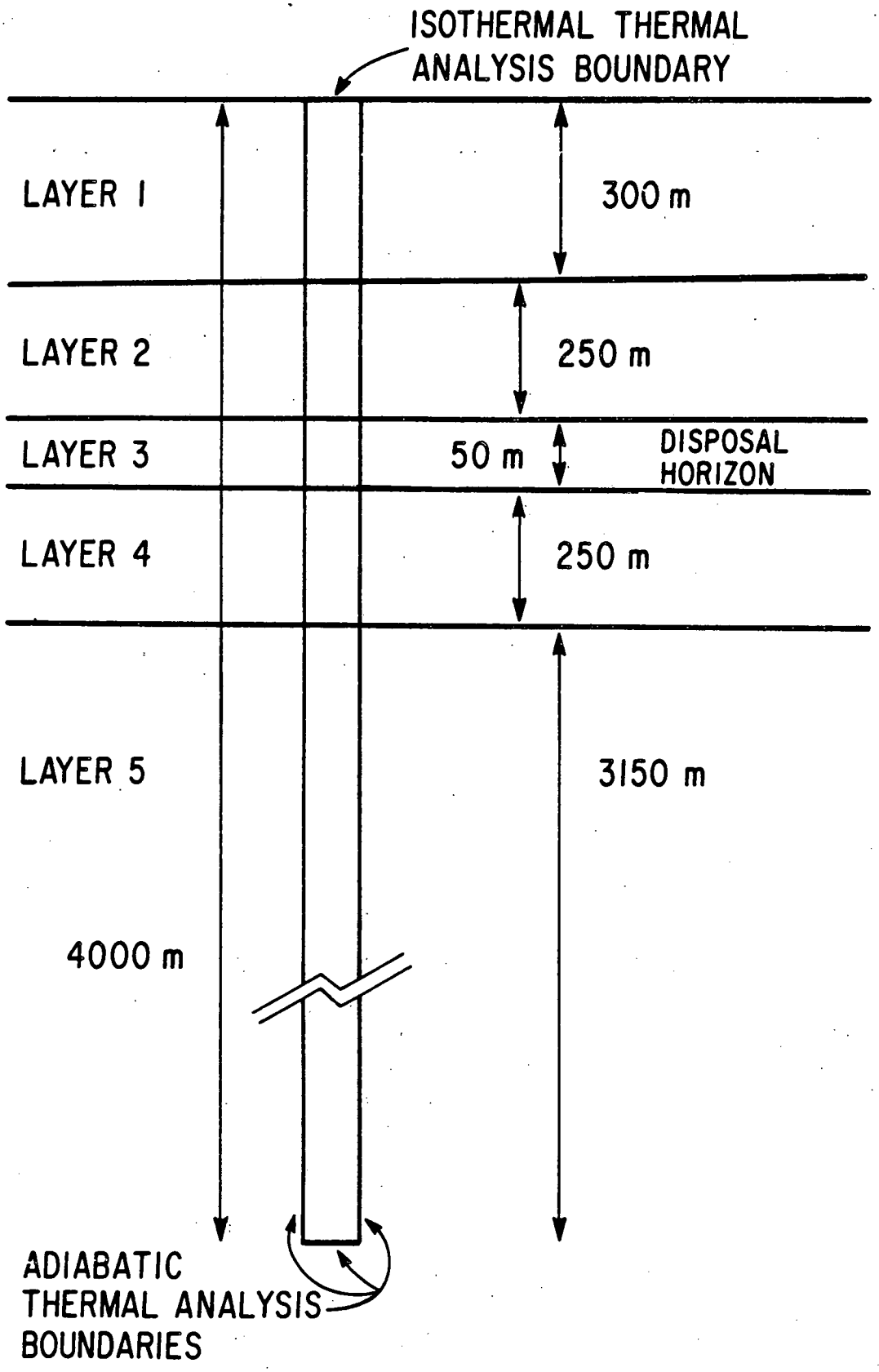


Fig. 3.1. Reference Site and Thermal Model Characteristics.

The five waste types considered are: (1) HLW ($UO_2$) - high-level waste resulting from LWR operation with U-only recycle, (2) SF ($UO_2$) - spent fuel from the U-only or no-recycle options, (3) HLW (MOX) - high-level waste resulting from LWR operation with third recycle mixed oxide fuel, (4) SF (MOX) - spent fuel from the same fuel cycle, and (5) SPK PU - spiked plutonium as a waste resulting from LWR operation in an equilibrium U-only recycle fuel cycle. The calculated thermal power and the time-integrated thermal energy release ten years after discharge for the five waste types are shown as functions of waste age in Figures 3.2 and 3.3. (The waste radionuclide inventories assumed for the five waste types are given in Appendix A).

Repository thermal design criteria are selected principally to ensure that the isolation capability of the disposal formation will be maintained. Secondary factors are operational constraints and economics. The thermal design criterion used in Reference 1 and summarized in this section is based on the maximum thermal energy (MTE) that would be stored in the geologic formations. Thermal energy would be added to the geologic formations by the radioactive decay heat from the wastes. For the assumptions used in this calculation (vertical heat flow only, no aquifers present, etc.) the only way heat would leave the geologic formations would be through the surface. Hence MTE would occur when the heat flux leaving the surface equaled the heat flux due to waste emplacement. The heat flux from the waste and the heat flux at the surface for HLW ($UO_2$) and SF ($UO_2$) wastes is shown in Figures 3.4 and 3.5. The time after emplacement for MTE is different for these two waste types. Similar plots for the other waste forms are given in Reference 1.

The thermal energy added to the geologic formations by the waste can be related to potential physical displacements and strains induced within or between strata surrounding the repository. These displacements and strains could lead to the creation of water pathways through overlying formations of low permeability. If this were to occur, it would represent one event in a sequence which could lead to a release of radionuclides from the repository.

The MTE stored in the geologic media would depend on: (1) waste emplacement density, (2) waste type, (3) dimensions and thermal properties of the individual strata, (4) emplacement depth from the surface, and (5) the presence of thermal sinks, such as aquifers. For the calculations described here, it has been assumed that (1) no thermal sinks are present, (2) the reference site conditions (dimensions, thermal properties, disposal depth) are as defined in Table 3.1 and Figure 3.1, and (3) the value of waste emplacement density for each type of waste (ten years old) is such that the MTE is equivalent to that for ten-year-old HLW ($UO_2$) emplaced at 106 kW/acre. This HLW ($UO_2$) emplacement density was chosen because it is more conservative (in terms of disposal area requirements) than the 150 kW/acre value given in Reference 11 and in NUREG-0002 (GESMO) and NUREG-0116 (S-3) area estimates and because it is used in a recent description of a bedded salt repository.*[12]

The thermal model used to calculate the MTE for each waste type employed a one-dimensional finite difference heat transfer code[13] simulating the thermal response to a 5-meter (15-ft) thick homogenous layer heat source. The calculated thermal response to a homogenous layer heat source includes all of the energy released by the waste, but does not include the two-dimensional thermal gradients near waste canisters. Vertical temperature profiles, surface heat fluxes, and the thermal energy stored in a 1-$m^2$ cross-section column [extending from the surface through the disposal horizon at 575 meters (1890 ft) to a maximum depth of 4000 meters (13,000 ft)--see Fig. 3.1] have been calculated for all waste types at emplacement densities defined by equivalent MTE. These initial emplacement densities and the area requirement ratios between the waste types and HLW ($UO_2$) are given in Table 3.2. Selected results of the thermal analyses for HLW ($UO_2$) and SF ($UO_2$) are presented in Figures 3.4 thru 3.10. A more detailed discussion of assumptions, techniques, and results is available in Reference 1.

In Figure 3.6 the average disposal horizon temperature rise over ambient is shown as a function of time after emplacement for HLW ($UO_2$) and SF ($UO_2$). These average temperature rises do not include short-term, near-field, two-dimensional gradients near waste canisters, and correspond to the emplacement densities indicated in Table 3.2 for the respective waste types. The energy content of each geologic layer as a function of time after emplacement is shown in Figures 3.7 and 3.8. Since the vertical distance between lines indicates the energy content of each layer, the total energy content is indicated by the height of the top line. The maximum value for each waste type is $3.34 \times 10^{10}$ J/$m^2$, indicating application of the maximum thermal energy criterion used to calculate relative emplacement density. For both waste types, energy first is deposited in layer 3, then diffused into layers 2 and 4, then after a few hundred years, into layers 1 and 5. Vertical temperature profiles at $10^1$, $10^2$, $10^3$, and $10^4$ years after emplacement are indicated in Figures 3.9 and 3.10.

*32 canisters/room and gross room dimensions of 78 ft by 590 ft (page III-2 of Reference 12), combined with 3.5 kW/canister (page i of Reference 12) yields 106 kW/acre.

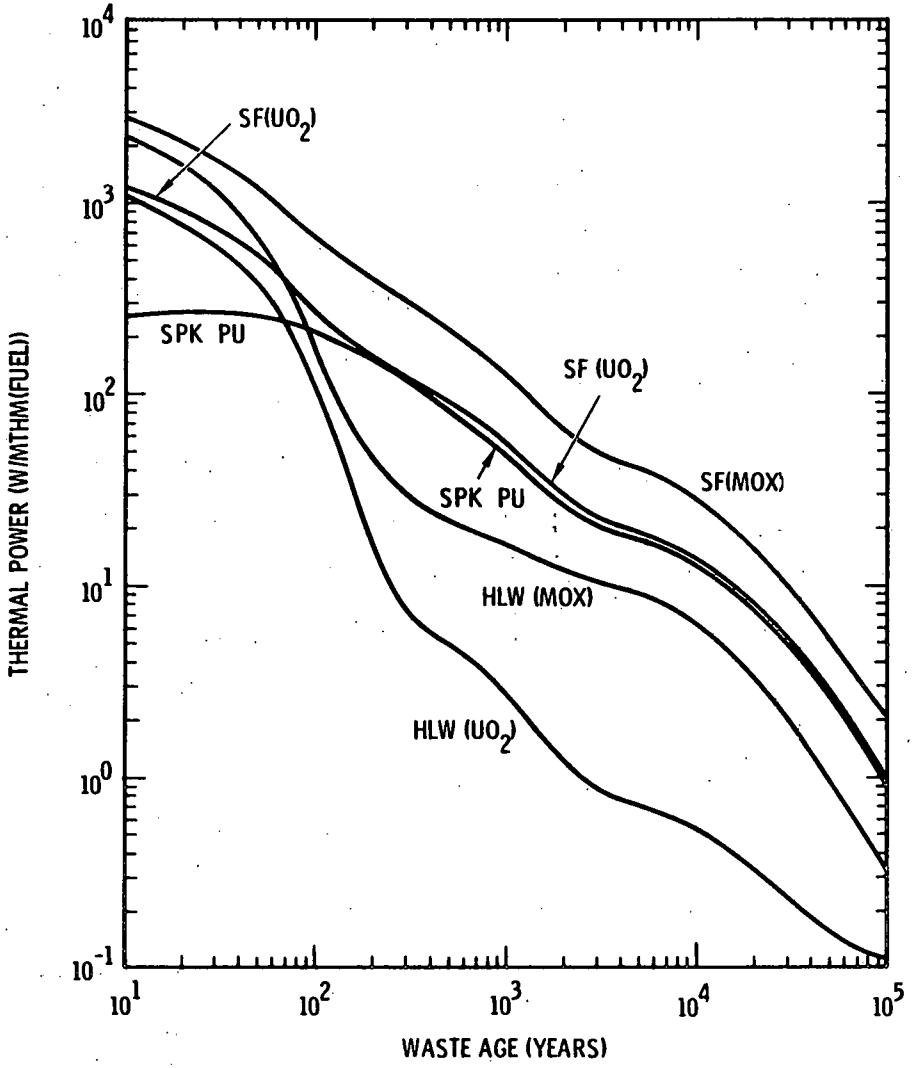


Fig. 3.2. Thermal Power of the Five Waste Types as a Function of Waste Age.

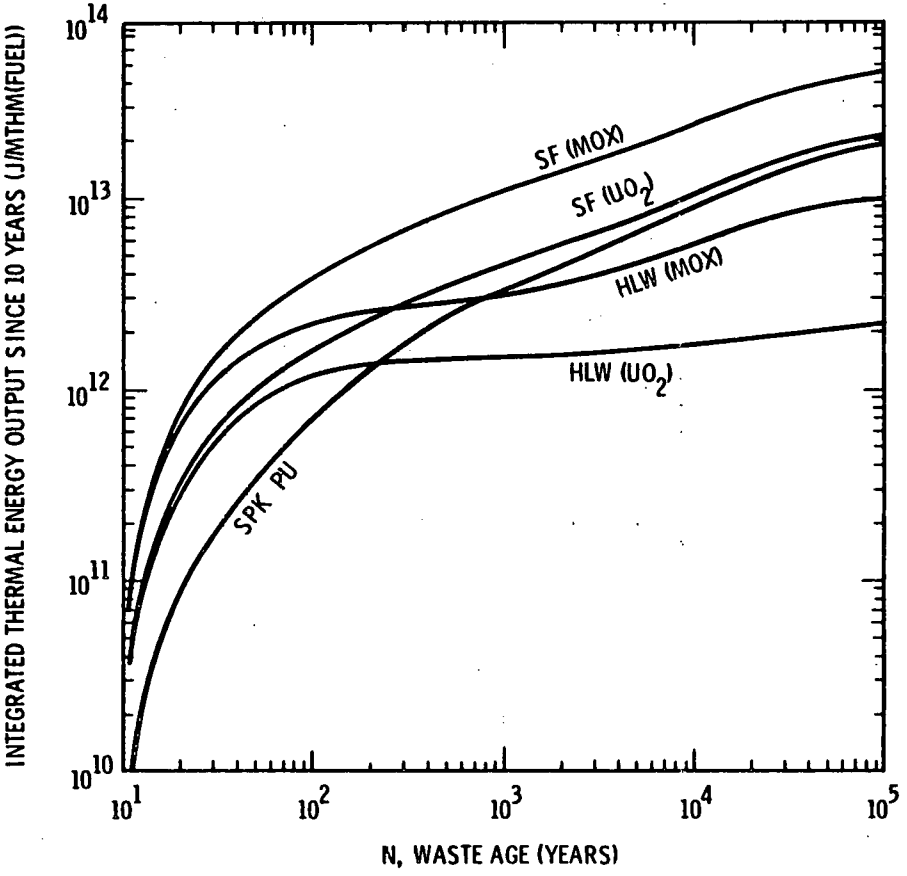


Fig. 3.3. Integrated (10 years to "N" years) Thermal Energy Output of the Five Waste Types.

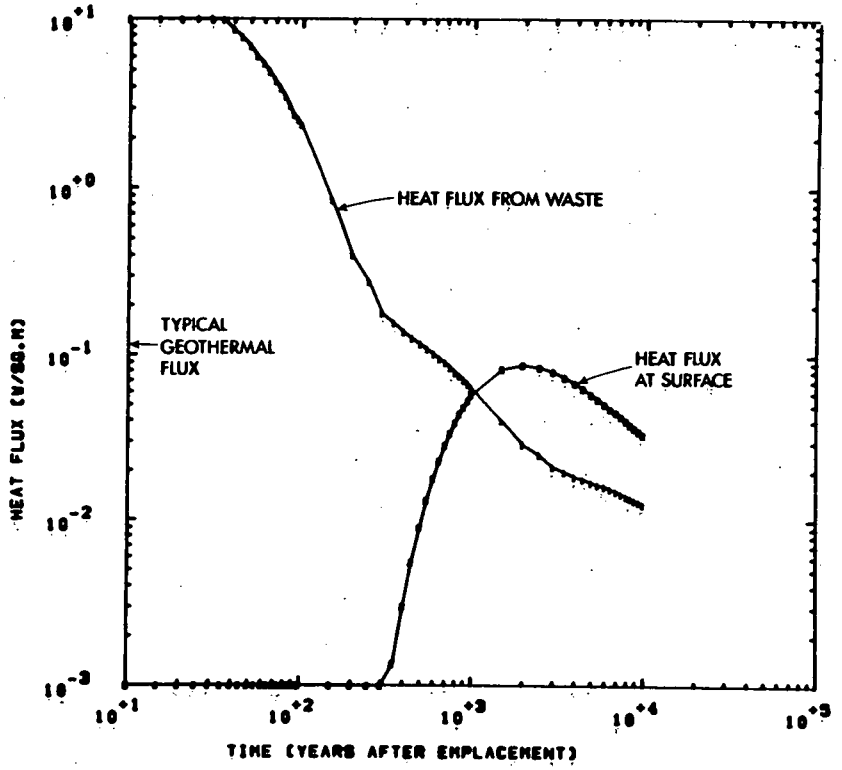


Fig. 3.4. Heat Flux from Waste and Heat Flux at Surface for HLW ($UO_2$) Emplaced at 106 kW/Acre.

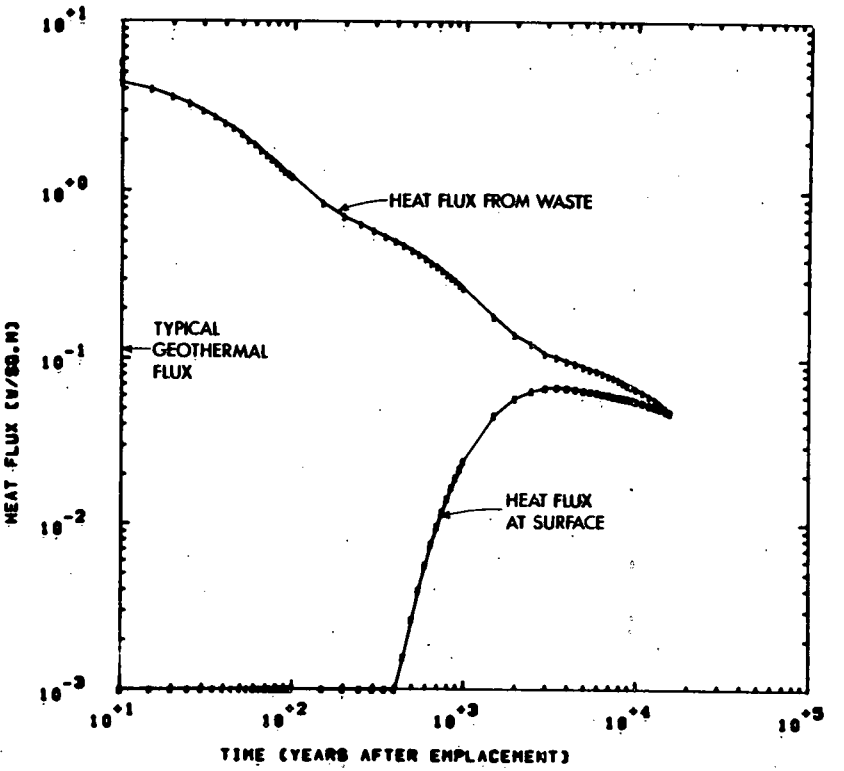


Fig. 3.5. Heat Flux from Waste and Heat Flux at Surface for SF ($UO_2$) Emplaced at 23.5 kW/Acre.

Table 3.2. Emplacement Density Characteristics for the Five Waste Types in the Reference Site

| Waste Type | Thermal Power of Waste at Assumed Emplacement Time[a] kW/MTHM(fuel) | Emplacement Thermal Flux for Equivalent MTE,[b] kW/acre | Approximate Time After Emplacement for MTE, years | Estimated Emplacement Density,[b] MTHM(fuel) /acre | Estimated Disposal Area,[b] $m^2$/MTHM (fuel) | Estimated Disposal Area Ratio, $m^2$/MTHM(fuel) / $m^2$/MTHM(fuel)-HLW ($UO_2$) |
|---|---|---|---|---|---|---|
| HLW ($UO_2$) | 1.11 | 106 | 1,000 | 95 | 42 | 1.0 |
| SF ($UO_2$) | 1.21 | 23.5 | 15,000 | 19 | 210 | 4.9 |
| SPK PU | 0.256 | 5.68 | 18,000 | 22 | 180 | 4.3 |
| HLW (MOX) | 2.29 | 85.4 | 12,000 | 37 | 110 | 2.6 |
| SF (MOX) | 2.78 | 24.2 | 14,000 | 8.7 | 470 | 11 |

[a] HLW ($UO_2$), SPK-PU, and HLW (MOX) are assumed to be emplaced ten years after reprocessing, which is assumed to occur 160 days out of core. SF ($UO_2$) and SF (MOX) are also assumed to be emplaced ten years and 160 days out of core.

[b] The values in these columns scale with the assumed 106-kW/acre initial emplacement thermal flux for HLW ($UO_2$). The relative values are independent of this assumption, as are the values in the estimated disposal area ratio column. All of the values are subject to the assumptions discussed in this text and in Reference 1.

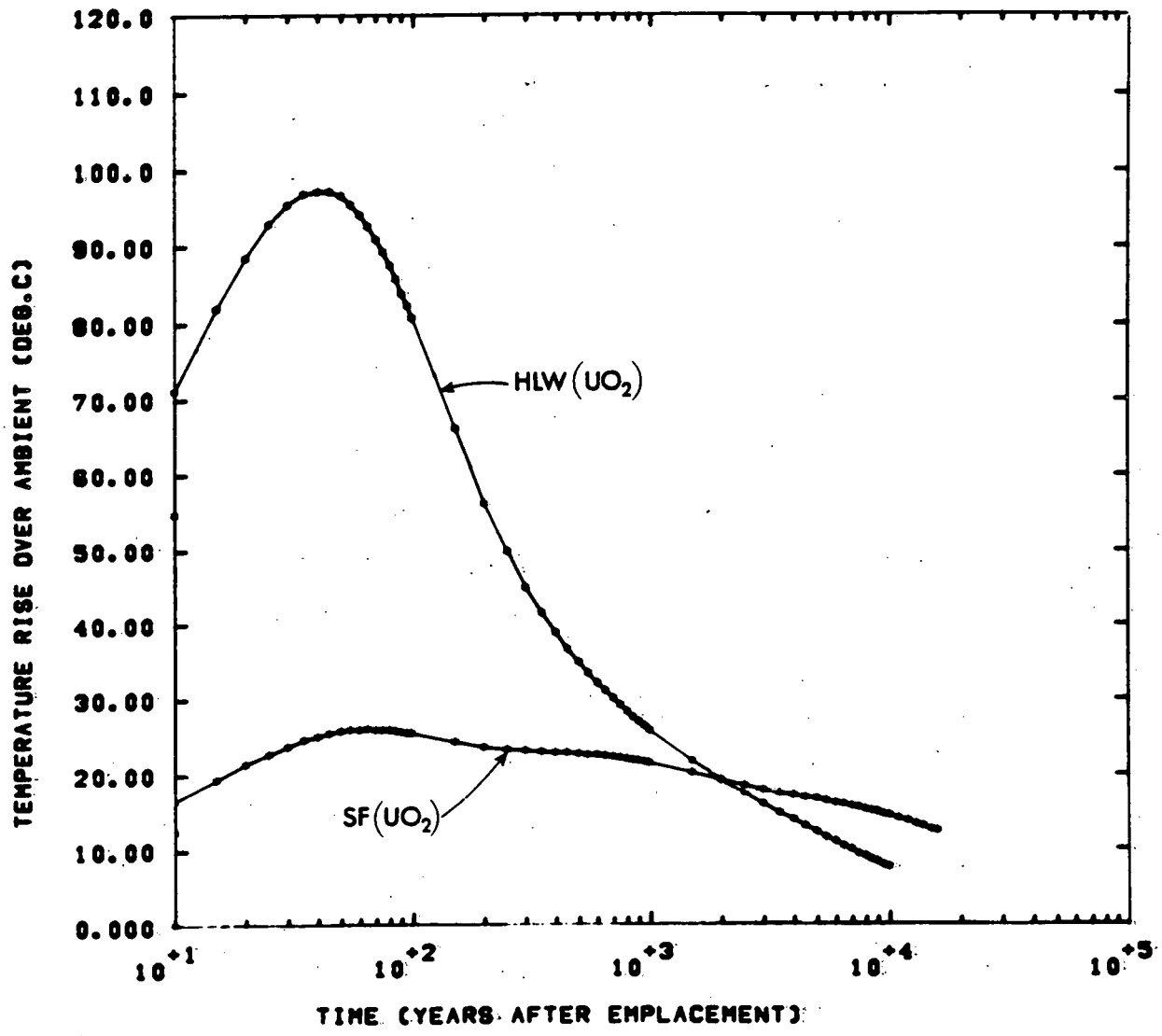


Fig. 3.6. Average Temperature Rise at Waste Disposal Level for HLW ($UO_2$) Emplaced at 106 kW/Acre and SF ($UO_2$) Emplaced at 23.5 kW/Acre.

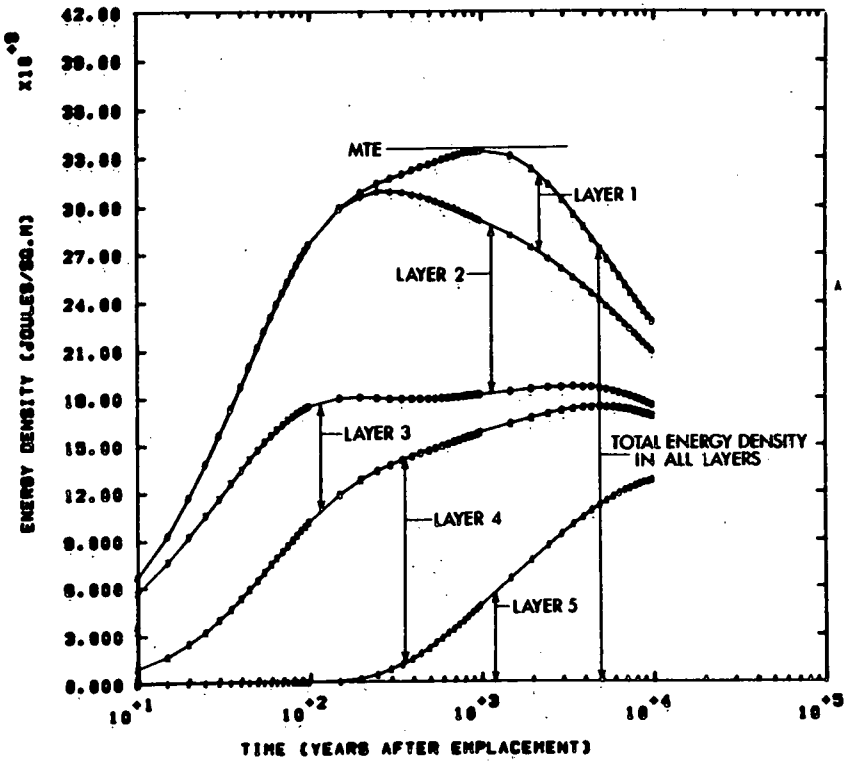


Fig. 3.7. Energy Stored in Geologic Layers for HLW ($UO_2$) Emplaced at 106 kW/Acre.

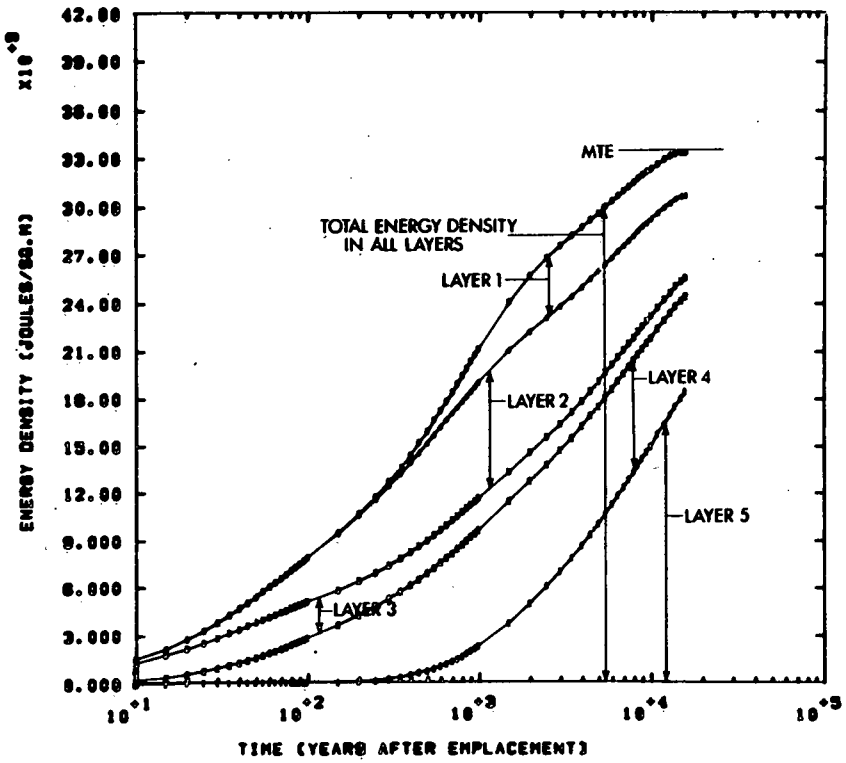


Fig. 3.8. Energy Stored in Geologic Layers for SF ($UO_2$) Emplaced at 23.5 kW/Acre.

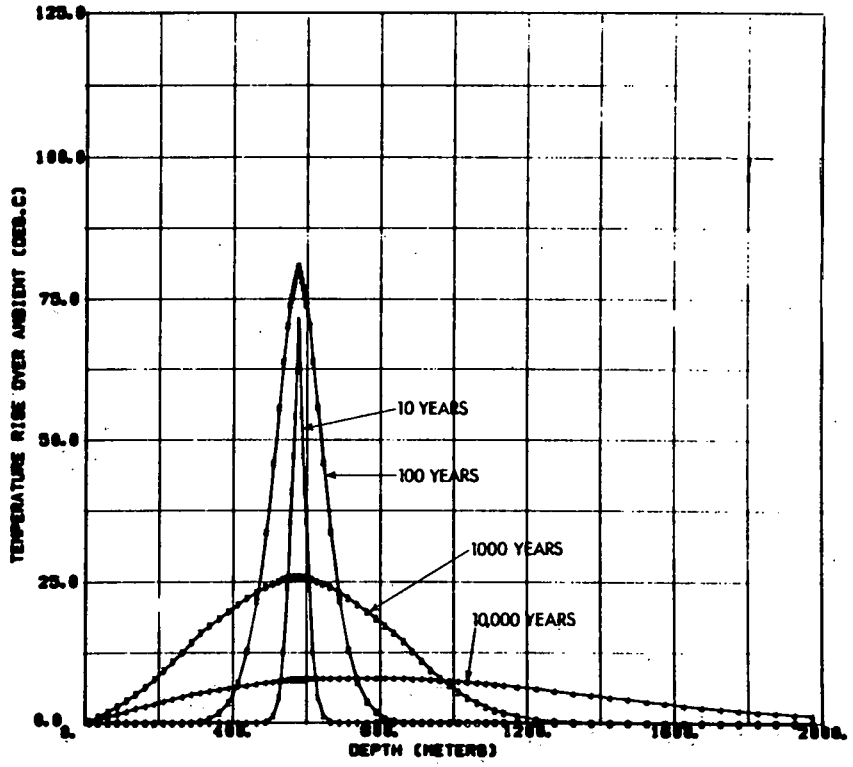


Fig. 3.9. Vertical Temperature Profiles for HLW ($UO_2$) Emplaced at 106 kW/Acre.

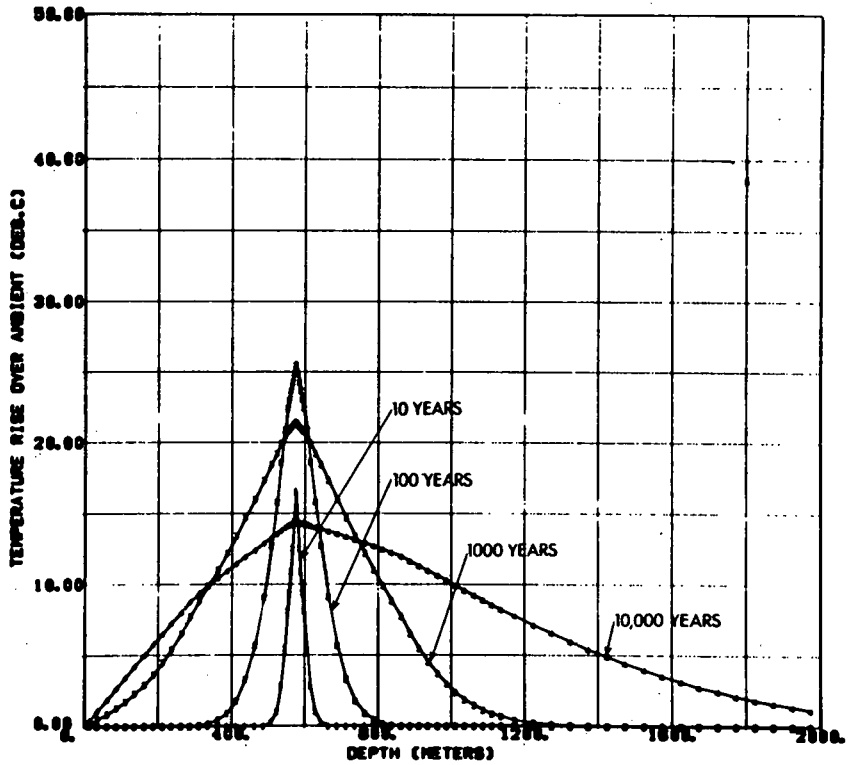


Fig. 3.10. Vertical Temperature Profiles for SF ($UO_2$) Emplaced at 23.5 kW/Acre.

The homogenized layer thermal source used in the analysis does not simulate short-term, near-canister thermal gradients. Near-canister gradients for SF ($UO_2$), SPK-PU, and SF (MOX) would generally be smaller than for HLW ($UO_2$) and hence would not limit emplacement density. However, near-field effects might control the canister size, the geometry, or the emplacement density for HLW (MOX) waste since this waste has about two times the specific power of HLW ($UO_2$) waste [2.29 versus 1.11 kW/MTHM (fuel)]. These calculations for relative emplacement density of various waste types are only approximate. For instance, the rate and amount of subsidence are not taken into account since these factors would depend on mine design and emplacement technique. The MTE criterion also does not take into account operational constraints (retrievability times, working temperature limits, etc.) on emplacement density. In some ways the criterion and calculational procedures are conservative. The equivalent MTE criterion includes the assumption that HLW ($UO_2$) emplacement density is limited by an allowable MTE. This limit is site specific. If the emplacement density of HLW ($UO_2$) for a specific site is not MTE-limited, the allowable emplacement densities of other waste types might be higher, and the estimated disposal area ratios presented in Table 3.2 lower. The longer-lived thermal output waste forms would produce more radial heat diffusion from the repository, thus this one-dimensional calculation is conservative. The MTE for long-lived wastes peak when much of the energy is in layers 1 and 5, which have a much lower volumetric expansion coefficient than layers 2, 3, and 4 ($27 \times 10^{-6}$/°C for 1 and 5, $73.5 \times 10^{-6}$/°C for 2 and 4, $120 \times 10^{-6}$/°C for 3).[9] Hence, the maximum surface uplift is less for longer-lived thermal output waste types when emplacement densities are calculated using the MTE criterion. Since the time to MTE is longer, the geologic strain rates are also lower for long-lived thermal output wastes. The lower strain rates might allow creep mechanisms to absorb larger total strains without creating potential water pathways. Rate-dependent phenomena, such as creep, have not been included in the present analysis.

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# 4. RADIOLOGICAL IMPACTS

## 4.1 INTRODUCTION

Individuals could be exposed to radiation as a result of normal operations or accidents at waste repositories. The exposed individuals could be workers at the repositories (receiving occupational exposure) or members of the general population (receiving the population exposure). Calculations of doses to both groups for normal operation and doses to the general population for accidents are described in this section.

## 4.2 ASSUMPTIONS AND PARAMETERS

Principal assumptions and parameters used in the analysis of radiological impacts are outlined in Table 4.1. They are based on current technology and on the extensive literature concerning design and operating experience of existing fuel and waste handling and treatment facilities. In cases where necessary information could not be obtained from experience at operating facilities, predictions were made on the basis of information available for projected facilities. Dose estimates were adjusted to apply over the period 1980-2140, with allowances made for operational occurrences and for plant aging effects over this time period. The year 2140 was used as the endpoint for these estimates to allow for a 100-year observation period following respository closure.

In treating dispersions and effects, equilibration between geosphere, hydrosphere, and atmosphere (e.g., resuspension of terrestrial radioactivity, aqueous deposition of atmospheric species, migration via ocean current and groundwater), as well as among various trophic levels within the biosphere, was considered. Such considerations were particularly important because many of the radioactive species, although produced at relatively constant rates, also tend to decay and to exhibit various other forms of time dependency in their equilibration with the environment.

Without leaks in the containment barriers there would be no release of radioactivity to the environment. Thus, on analysis, the primary consideration in minimizing such releases would be the quality control of barrier integrity, both short and long term. For purposes of assessment it was assumed that each waste form was contained by one intact barrier before entry into the surge pool and by two intact barriers before entry into its intended repository facility. Without such containment, even the most extensive system of subsequent restraints (e.g., multiple HEPA filters, scrubbers) would be unable to maintain releases at an acceptably low level.

It was assumed that reprocessing operations would begin in 1982, and that initial operations would begin with the backlog of spent fuel available at that time. All radioactive material would have been aged ten years before receipt at a repository. The fuel production schedule assumed herein is based on an installed nuclear generating capacity rising to 507 GWe in 2000, with new plants being added both to increase capacity and to replace retired plants. Installation of new capacity was assumed to cease at the year 2000, and the amount of fuel being discharged would drop as plants reached the end of their operating lifetime, with the last plant closing in 2030. Waste would continue to move through the repositories until 2040 (see Sec. 5).

## 4.3 WASTE AND FACILITY DESCRIPTIONS

The various types of wastes that could be handled at a repository, as described in Section 2, are spent fuel assembiles (SF); high-level solidified waste (HLSW); fuel bundle residues (hulls); transuranic intermediate-level waste (TRU-ILW); transuranic low-level waste (TRU-LLW); and spiked $PuO_2$. The important properties of these wastes relevant to radiological impact are summarized in Table 4.2.

All of the waste types were taken into account for calculation of the doses from normal operations. For the accident analysis, only the spent fuel and high-level solidified wastes were considered since the impacts from accidents involving these two types of waste would be much more significant than for the other types.

Table 4.1. Principal Assumptions and Parameters Used in Computation of Releases and Doses

| | |
|---|---|
| **Fuel Basis:** | |
| Burnup | $3.3 \times 10^4$ MWd/MTHM |
| Specific power | 30 MW/MTHM |
| Reactor mix, PWR:BWR | 2:1 |
| Leakage coefficient, each barrier | $1 \times 10^{-3}$/yr |
| Leakage coefficient, each fuel rod | $1 \times 10^{-5}$/yr |
| Cooling time, to arrival | 10 yrs |
| $PuO_2$ fission-product addition | 5% (by weight) |
| **Waste Treatment and Dispersion:** | |
| Noble gas, H and C transmission | 1 |
| Iodine transmission | $1 \times 10^{-3}$ |
| Semivolatile transmission | $1 \times 10^{-7}$ |
| Particulate transmission | $1 \times 10^{-9}$ |
| Stack height | $1 \times 10^2$ m |
| Stack flow | $1 \times 10^2$ m$^3$/sec |
| Stack velocity | 8 m/sec |
| Stack exhaust temperature, above ambient | 5 deg K |
| Center to controlling site boundary | $2 \times 10^3$ m |
| Initial specific power | $9 \times 10^3$ kW/m$^2$ |
| **Surge Pool Basis:** | |
| Activity, total | $1 \times 10^{-3}$ Ci/m$^3$ |
| Activity, composition | 60% Cs-137/134, 25% Co-60/58, 9% H-3, 5% Ni-63, 1% all other $\beta$-$\gamma$, 0.01% alpha |
| In-flow | $3 \times 10^2$ liter/day |
| Filter flow | $4 \times 10^3$ liter/min |
| Cation/anion exchanger flow | $2 \times 10^3$ liter/min |
| Heat load | 1 MW |
| Mean fuel load | $8 \times 10^2$ MTHM |
| **Repository Operations (as required by alternative):** | |
| Maximum fuel throughput | 30 assemblies/day |
| Maximum high-level waste throughput | 3 canisters/day |
| Maximum $PuO_2$ throughput | 20 canisters/day |
| Mean 8-hour workdays | $1 \times 10^3$/yr |
| Mean employment, 1982-2040 | 500/repository |
| Mean employment, 2041-2141 | 20/repository |
| **Demography:** | |
| Low population zone (LPZ), uniform density | 5/km$^2$ |
| LPZ to 80 km, uniform density | 100/km$^2$ |
| Population characteristics | U.S. Census 1970 |

A flow diagram of the basic steps involved in the receipt, handling, and emplacement of the waste at a waste respository is presented in Figure 4.1. In performing the radiological analyses, the staff assumed that the basic facilities required for each option would be colocated and interconnected to facilitate waste handling and to minimize the chance of accidents. (Releases during transport of the wastes to the facility were not considered). As shown in the figure, there are four basic types of facilities:

(1) Surge pool facility. Spent PWR and BWR fuel elements, high-level solidified waste (HLSW), and other canistered wastes would be received here for interim storage. This facility would be similar for both the no-recycle and the recycle alternatives, with the primary difference between the two being the type of racks placed in the pool for holding either spent fuel assemblies or canistered wastes. It is assumed that a reference surge pool would be 76.2 m (250 ft) long by 18.3 m (60 ft) wide and filled to a depth of 12 m (40 ft) with $1.7 \times 10^7$ liters (4.4 million gallons) of water. It is further assumed that the building housing the pool and receiving facilities would be maintained at a negative pressure relative to atmospheric and that the air would be exhausted through two stages of HEPA filtration.

Table 4.2. Properties of Waste Streams for Radiological Impact Analysis

| Waste Form | Waste Volume per Container, $m^3$ | Waste Weight per Container, kg | Gross Ci Content | Heat Generated, W | Location in Fig. 4.1 |
|---|---|---|---|---|---|
| PWR SF[a] ($UO_2$) | 0.20 | 600 | 179,000 | 545 | A,B,C[b] |
| PWR SF (MOX) | 0.20 | 600 | 248,000 | 1,260 | A,B,C[b] |
| BWR SF ($UO_2$) | 0.10 | 250 | 79,600 | 242 | A,B,C[b] |
| BWR SF (MOX) | 0.10 | 250 | 110,000 | 556 | A,B,C[b] |
| HLSW ($UO_2$) | 0.177 | 390 | 980,000 | 3,450 | D,E,F[c] |
| HLSW (MOX) | 0.177 | 390 | 1,090,000 | 7,160 | D,E,F[c] |
| $PuO_2$ | 0.003 | 6 | 45,700 | 153 | G |
| Hulls | 0.177 | 177 | 1,560 | 15.9 | D,E,F[c] |
| TRU-LLW | 0.167 | 167 | 12.2 | 0.0339 | G |
| TRU-ILW | 0.177 | 177 | 7.31 | 0.0343 | D,E,F[c] |

[a]SF = spent fuel (received in intact spent fuel assemblies).

[b]At C in Figure 4.1, the assemblies are in canisters; at A & B, they are uncanistered.

[c]At F in Figure 4.1 the canisters are contained in overpacks; at D & E they are not overpacked.

(2) Encapsulation/overpack facility. This would be a shielded hot cell in which spent fuel assemblies and canistered wastes would be overpacked when they were received at the facility. All transfers of spent fuel or canistered waste from the receiving casks to the surge pool and from the pool to the encapsulation/overpack cell would be performed underwater via transfer canals. It has been assumed that the air atmosphere in the hot cells would be maintained at a negative pressure and that all exhausted air would pass through two stages of HEPA filtration. Upon receipt, leaking waste containers would be sent directly to the encapsulation/overpack cell, where they would be doubly canistered and then sent to the emplacement area.

(3) Caisson surface storage facility. This facility would be used for interim storage of canistered spent fuel assemblies.

(4) Underground deep mine storage/disposal. All wastes, except any spent fuel stored in the caisson surface facility, would be placed in underground deep mine facilities for storage/disposal.

## 4.4 NORMAL OPERATIONAL RELEASES

Population and occupational doses associated with each fuel cycle option under normal operating conditions are summarized in Table 4.3. The doses given are upper bounds and are the sums of those to the most critical organs or tissue for radionuclides or radiations involved. In the case of the no-recycle options, the upper bound proved to be the dose to the skin from Kr-85. For the various recycle options, the critical doses were about equally divided between bone (Sr-90) and lung (plutonium and transplutonium nuclides). Occupational doses were invariably dominated by direct radiation, with genetically significant dose as the determinant. The values given in Table 4.3 are average annual doses for the operational period (through 2040) and for the postoperational century of repository management (2041-2141). Values given are for the population within 80 km (50 miles) of each repository and for workers at the repositories summed for all repositories. Background values are included for comparison and are given for a mean natural dose rate of 0.1 rem per year.

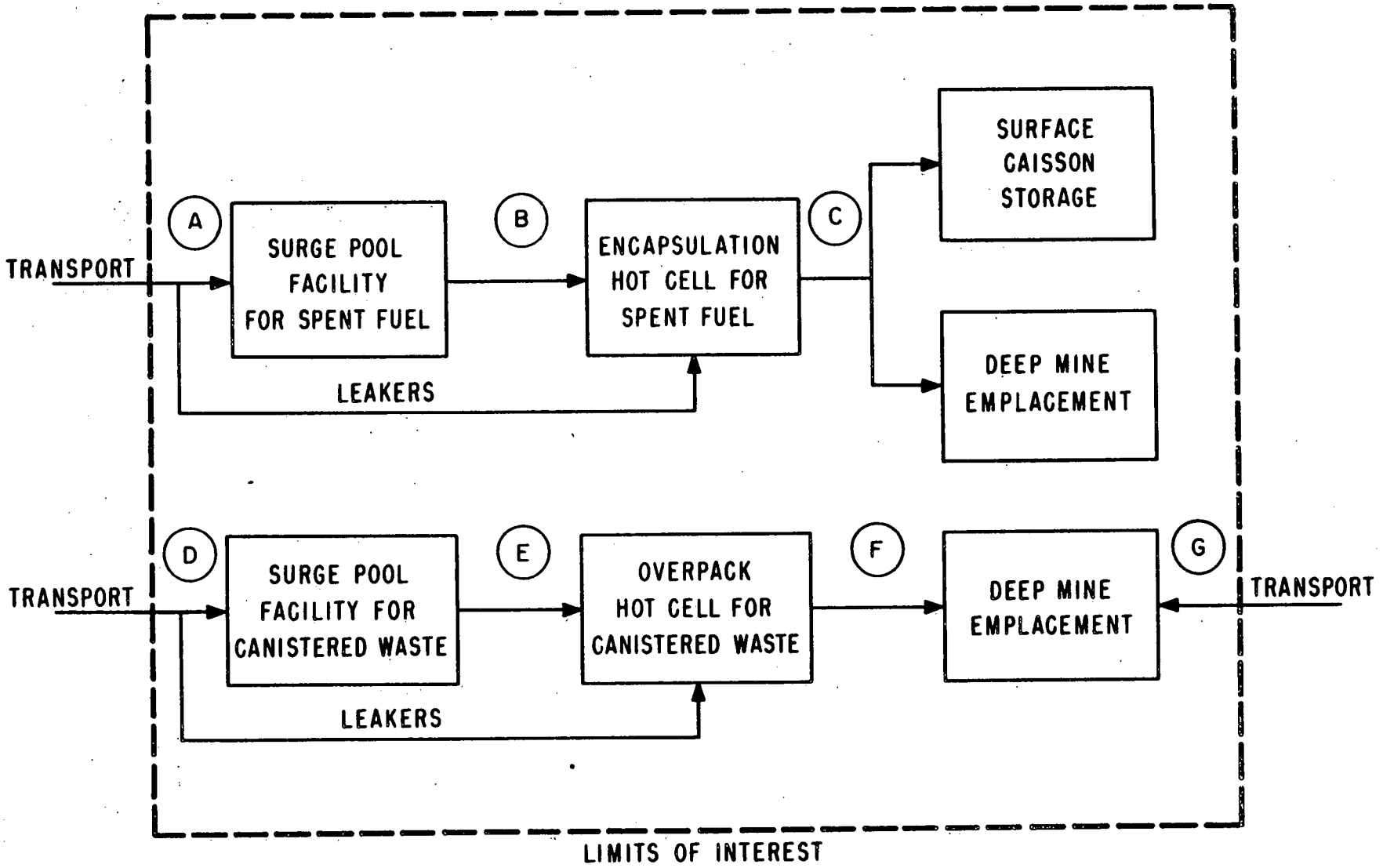


Fig. 4.1. Flow Diagram for Waste Streams in Waste Management Radiological Impact Analysis. (See Table 4.2 for definition of waste streams.)

Table 4.3. Radiological Doses Associated with Waste Management Options, Normal Operating Conditions

| Fuel Cycle Option | Population Dose, man-rem/yr[a] | | Occupational Dose, man-rem/yr[a] | |
|---|---|---|---|---|
| | Through 2040 | 2041-2141 | Through 2040 | 2041-2141 |
| U-recycle, Pu stored | $4 \times 10^{-3}$ | $3 \times 10^{-1}$[b] | $4 \times 10^{3}$ | 40[b] |
| U-recycle, Pu disposed | $3 \times 10^{-3}$ | $1 \times 10^{-1}$ | $3 \times 10^{3}$ | 30 |
| Full recycle | $5 \times 10^{-3}$ | $1 \times 10^{-2}$ | $8 \times 10^{2}$ | 6 |
| No-recycle, surface storage | 30 | 7[c] | $4 \times 10^{3}$ | 40[c] |
| No-recycle, deep stowaway | 4 | $5 \times 10^{-1}$[b] | $3 \times 10^{3}$ | 10[b] |
| No-recycle, deep throwaway | 1 | $4 \times 10^{-1}$ | $1 \times 10^{3}$ | 10 |
| Background, 9 repositories | $6 \times 10^{5}$ | $6 \times 10^{5}$ | $2 \times 10^{2}$ | 6 |
| Background, 14 respositories | $1 \times 10^{6}$ | $1 \times 10^{6}$ | $3 \times 10^{2}$ | 10 |

[a]Averaged values over the time period involved.

[b]For the two retrievable storage options, it is assumed that the storage rooms would be backfilled after an interim period. The doses shown here do not account for any shielding effects of this backfill.

[c]For the surface storage option, the spent fuel assemblies ultimately would be disposed of following this interim storage period. The doses shown here are based on the assumption that the spent fuel assemblies would be left in the surface storage facilities until 2140. These doses are given for comparative purposes only.

Present regulations set maximum permissible doses at 0.5 rem per year to any member of the public and 5 rem per year to any employee; however, current experience indicates actual values of less than 0.005 and 1.5 rem/yr, respectively, for facilities of this kind. For future facilities these values can be expected to be reduced even further. Thus, it is estimated that for each option, the overall dose to the public would be many orders of magnitude below that resulting from the natural background, and the dose to the workers (occupational dose) would be within an order of magnitude above background.

## 4.5 ACCIDENTAL RELEASES

In this section, radioactive materials available for release are listed, potential accidents are described and assigned probabilities, and releases and doses are calculated for the most likely accidents.

### 4.5.1 Source Terms

All waste forms are assumed to have aged ten years before receipt at the waste repository. The nature and magnitude of the radioactivity available for release vary considerably over the spectrum of waste forms. The forms containing fission products are most radioactive, as shown in Table 4.4.

### 4.5.2 Accident Descriptions

Nine potential accidents that could result in releases of significant amounts of radioactivity at a waste repository are analyzed in the following sections.

#### 4.5.2.1 Container Drop Accident

The likelihood of a container drop accident would depend upon the handling times and procedures, and the consequences would depend on the form of the waste. All waste forms would enter the repository with at least one intact containment layer. Wastes containing fission products would be stored or disposed only after being surrounded by two nonleaking containment barriers. This defense-in-depth philosophy would provide protection from leaking containers and provide strength to maintain containment integrity for all but the most severe shocks and blows.

Table 4.4. Source Terms for Ten-Year-Old Waste Forms Used in Accident Calculations[a]

| Radionuclide | Spent Fuel (Ci/MTHM) | | High-Level Waste (Ci/canister) | | Spiked $PuO_2$ (Ci/container) |
|---|---|---|---|---|---|
| | $UO_2$ | MOX | $UO_2$ | MOX | |
| Gases | | | | | |
| H-3 | $4.04 \times 10^2$ | $4.04 \times 10^2$ | -- | -- | -- |
| Kr-85 | $5.95 \times 10^3$ | $5.95 \times 10^3$ | -- | -- | -- |
| I-129 | $3.71 \times 10^{-2}$ | $3.71 \times 10^{-2}$ | -- | -- | -- |
| Total | $6.35 \times 10^3$ | $6.35 \times 10^3$ | | | |
| Volatiles | | | | | |
| Cs-134 | $7.82 \times 10^3$ | $7.82 \times 10^3$ | $2.45 \times 10^4$ | $2.45 \times 10^4$ | $6.84 \times 10^1$ |
| Cs-137 | $8.57 \times 10^4$ | $8.57 \times 10^4$ | $2.68 \times 10^5$ | $2.68 \times 10^5$ | $7.50 \times 10^2$ |
| Total | $9.35 \times 10^4$ | $9.35 \times 10^4$ | $2.92 \times 10^5$ | $2.92 \times 10^5$ | $8.18 \times 10^2$ |
| Particulates | | | | | |
| Sr-90 | $5.95 \times 10^4$ | $5.95 \times 10^4$ | $1.86 \times 10^5$ | $1.86 \times 10^5$ | $5.20 \times 10^2$ |
| Y-90 | $5.96 \times 10^4$ | $5.96 \times 10^4$ | $1.86 \times 10^5$ | $1.86 \times 10^5$ | $5.21 \times 10^2$ |
| Pu-238 | $5.27 \times 10^2$ | $1.02 \times 10^4$ | $2.71 \times 10^2$ | $2.81 \times 10^3$ | $3.05 \times 10^3$ |
| Pu-239 | $3.24 \times 10^2$ | $4.69 \times 10^2$ | 5.10 | $2.20 \times 10^1$ | $1.90 \times 10^2$ |
| Pu-240 | $4.83 \times 10^2$ | $1.05 \times 10^3$ | $1.28 \times 10^1$ | $1.47 \times 10^2$ | $2.84 \times 10^2$ |
| Pu-241 | $7.71 \times 10^4$ | $1.84 \times 10^5$ | $1.20 \times 10^3$ | $2.89 \times 10^3$ | $3.87 \times 10^4$ |
| Pu-242 | 1.74 | 7.91 | -- | -- | $8.16 \times 10^{-1}$ |
| Am-241 | $1.80 \times 10^3$ | $4.24 \times 10^3$ | $7.57 \times 10^2$ | $1.65 \times 10^3$ | $7.86 \times 10^2$ |
| Am-243 | $1.73 \times 10^1$ | $2.14 \times 10^2$ | $5.41 \times 10^1$ | $6.70 \times 10^2$ | -- |
| Cm-242 | 5.70 | $7.27 \times 10^1$ | $1.78 \times 10^1$ | $2.28 \times 10^2$ | -- |
| Cm-244 | $1.34 \times 10^3$ | $3.37 \times 10^4$ | $4.19 \times 10^3$ | $1.05 \times 10^5$ | $1.48 \times 10^1$ |
| Total | $2.01 \times 10^5$ | $3.53 \times 10^5$ | $3.78 \times 10^5$ | $4.85 \times 10^5$ | $4.41 \times 10^4$ |

[a]The nuclides selected correspond with those considered "biologically significant" in Appendix 3 of "Determination of Performance Criteria for High-Level Solidified Nuclear Waste," Lawrence Livermore Laboratory, LLL-NUREG-1002, April 1977. Data have been extracted from Appendix A of this report.

It is assumed that within a repository, wastes containing fission products would be moved with cranes having a nominal drop probability of $3 \times 10^{-6}$ drops per hour of handling.[1] Handling time for each movement is assumed to be 20 minutes. An exception is that a 30-minute handling time is assumed for transport of canistered spent fuel assemblies within the caisson storage area.

Spent fuel assemblies might be dropped (1) during underwater removal from the shipping cask and transfer to the surge pool; (2) during transfer to the encapsulation area or while undergoing encapsulation; and (3) during transfer to surface or underground storage. The HLSW canisters also would be transferred through the surge pool and overpack facility and to underground disposal. (Releases from the drop of a shipping cask are not considered since Department of Transportation regulations require that the casks be able to withstand, without rupture, much more severe handling accidents than might occur in the waste facility.) Both the spent fuel and HLSW would be most vulnerable in the encapsulation/overpack cell since there they would be neither underwater nor within two layers of containment. Check mechanisms and test and maintenance programs for the elevator to the underground storage/disposal area are assumed sufficient to preclude any drop of the elevator or of its contents.

4.5.2.2 Drop of a Heavy Object on a Waste Container

Waste containers stored in surge pools or awaiting storage or disposal could be damaged if a heavy object fell on them; however, transfer cranes would be the only heavy machinery allowed over the surge pools, and it is assumed that adequate stops and checks would be built into the crane control system to preclude collisions of waste cansiters or inadvertent attempts to place canisters or assemblies in occupied spaces. The nominal probability of an object heavy enough to cause fuel cladding or HLSW canister rupture falling into the surge pool is assumed to be $10^{-4}$ per year.

4.5.2.3 Loss of Surge Pool Cooling

As seen in Table 4.2, waste forms containing fission products would generate substantial amounts of heat. After ten years of decay, individual spent fuel assemblies or HLSW canisters could be cooled sufficiently by free air convection, but in the close-packed surge pool configuration, cooling water would have to be continuously circulated to prevent overheating and rupture of the waste containers or assemblies. For analysis of this type of accident, it is postulated that (1) the pool cooling system fails; (2) because of incredible circumstances coolant flow cannot be reestablished; (3) the surge pool is filled to capacity; and (4) the only heat removal mechanisms available are heating and boiling of the surge pool water. If filled with $UO_2$ spent fuel assemblies, the noncirculating pool water would heat up at a rate of about 0.1° K/hr (0.2°F/hr) and would eventually boil at a rate of 3400 liters/hr (120 $ft^3$/hr). In about 80 days, the water level would reach the tops of spent fuel assemblies, causing some cladding failures.

The probabilities of the loss of cooling to a spent fuel storage pool at a reactor site and of subsequent failure of operators to recognize the need for makeup water are discussed in the Reactor Safety Study.[1] The nominal probability of failure of the pool-water cooling system is estimated at 0.1 per year. Failure to recognize the need for makeup water in the closed spent fuel pool for a period of several weeks after loss of cooling is estimated to occur with a probability of $10^{-6}$. However, the surge pool envisioned in this report would be under closer and more frequent inspection, as well as heavier instrumentation, than a spent fuel pool. Thus, it is assumed that once cooling had been interrupted, the probability of not being able to restore cooling by repairing the cooling system or adding makeup water from an alternative source would be an order of magnitude lower, or $10^{-7}$. Thus, the overall probability of releases due to coolant boiloff is $10^{-8}$/year.

4.5.2.4 Earthquake Greater than Safe Shutdown Earthquake (SSE)

The nuclear waste facility would be designed to Class 1 standards and would withstand an SSE without releasing radioactivity. Furthermore, the facility would be constructed in an area of minimum seismic activity. However, an earthquake more severe than the SSE could occur, causing surge pool drainage and overheating and rupture of the containers stored there. The surge pool building could also be ruptured. The probability of the occurance of an earthquake of this strength is taken as $10^{-5}$ per year, and it is assumed that there is a 0.1 probability that the severe earthquake would result in a pool drainage.[1] This would result in an overall probability of $10^{-6}$ per year.

4.5.2.5 Aircraft Impact

Radioactive materials could be released by the impact, and subsequent fire damage, of a large aircraft crashing into the waste facility. Only those wastes on the surface would be affected. Also, regular air routes and military training flights would be prohibited in the air space over the facility, making such an accident highly unlikely. The probability of an aircraft crashing into the surge pool is estimated to be $6.8 \times 10^{-14}$ per year.[2]

The probability of an aircraft's crashing into a spent fuel caisson storage area with sufficient impact to breach a stored assembly is calculated to be $3.7 \times 10^{-14}$ per caisson [based on a 7.6-m (25-ft) spacing between caissons].[2]

4.5.2.6 Fire or Explosion

The use and accumulation of combustible materials would be kept to a low level in all areas of the waste facility. The water cover of the surge pool and transfer canals and the multiple layers of containment around the waste forms would make it very unlikely that a fire or explosion would cause any release of radioactivity.

A potential explosion hazard inherent in the surge pool would be the generation of radiolytic hydrogen from the water. It has been estimated that the pool could generate about 0.001 $m^3/s$ (2.5 $ft^3/min$) of hydrogen.[3] The minimum hydrogen flammability limit in air is about four volume percent. Accordingly, if hydrogen were allowed to accumulate well above this limit in some portion of the building, an explosion could result. Therefore, the pool building would include both normal and Class 1 ventilation systems to ensure that the hydrogen concentrations were always well below the flammability limit.

#### 4.5.2.7 Tornado

The probability of a tornado strike is site dependent. A location conforming to NRC site criteria would have a low probability for a tornado. The waste facility structures would be designed to Class 1 requirements and, therefore, could withstand tornadoes. A metal-sided surge pool building, however, could be penetrated by tornado-generated missiles. For example, a tornado-generated missile--assumed to be a 0.3-m (12-inch) diameter by 6.1-m (20-ft) long pole weighing 286 kg (630 lb), traveling at 44.7 m/sec (100 mph)--directed vertically downward into the surge pool would be slowed down by the water so that it would only crack, not crush, one or more waste containers. If these containers were spent fuel assemblies, the gaseous fission products would be released and the surge pool water would become contaminated.

Tornado-generated missiles could also cause damage to the surge pool cooling system and, in particular, to the cooling towers on the secondary cooling circuit. The releases from such an accident would be similar to those for the loss of pool circulation without the ability to reestablish cooling. Since the probability of such an event is much lower than the normal system failure rate of $10^{-1}$ per year, the overall releases from a tornado-caused failure of the cooling system would be very much less than those for the loss of circulation accident. Furthermore, the very long time available (approximately 80 days before the coolant boiled off a surge pool filled with $UO_2$ spent fuel assemblies) would give ample time for repairing the damaged cooling system.

The probability of the tornado-caused event would be much lower than the probability of the heavy object drop accident, and since the consequences would be similar, the expected dose* would be much less.

#### 4.5.2.8 Flood

The waste facility would be located at an elevation satisfying NRC siting criteria; therefore, occurence of a flood that endangered the facility would be an incredible event. However, in cases where the surge pool cooling water supply was from a river or at an elevation susceptible to flood damage, an emergency water supply, such as a well, would be provided.

#### 4.5.2.9 Criticality

The considerations regarding nuclear criticality in the waste facility are essentially the same as those in fuel storage pools at reactor sites. This problem has been solved at the latter facilities by proper spacing of storage racks and, in some cases, by using racks containing neutron-absorbing materials. The same types of procedures would be used in the waste facility, thus making accidental criticality of fuel assemblies highly improbable, even in the face of gross human error.

### 4.5.3 Releases of Radioactivity

The amount of radioactive material released in an accident would depend on several variables.

(1) Contents of waste facility. For this analysis, the surge pool and other locations of the facility are assumed to be filled to capacity. A full surge pool is assumed to contain 1500 MTHM of spent fuel or 481 HLSW canisters.

(2) Radioactivity available in the waste form. The values assumed here have been presented in Section 4.5.1.

*The "expected dose" is defined in Section 4.5.4.

(3) Magnitude of damage. The severity of an accident and the effects on the waste canisters could vary greatly. For each accident analysis, an expected number of waste canisters affected is assumed. For example, for the drop of an uncanistered spent fuel assembly it is assumed that 20% of the fuel rods are breached. In contrast, the loss of circulation cooling accident might result in the rupture of all the waste containers in the surge pool.

(4) Escape mechanisms. The relative volatilities of the nuclides available for release would determine how much of each entered the building atmosphere or escaped the waste facility. Also, the accident environment--in air or underwater--would affect the release of volatile and particulate materials.

(5) Failure of other systems. Systems such as the building heating, ventilation, and air conditioning (HVAC) system, with its multiple filtration units, could be expected to reduce the releases from many of the accidents considered. In the specific case of the HVAC system, a probability of $10^{-6}$ was assumed for its failure during any accident which had not already damaged this system to render it inoperable.

Release factors for each type of radioactive material and for each stage of an accident were estimated on the basis of published information or assumptions by the staff. These release factors are presented in Table 4.5. Only the accidents and waste forms listed in the table were considered beyond this point, since other accidents were scoped by this set either because of the relatively high probabilities, as in the drop accidents, or because of the large amounts of radioactivity available, as in the whole pool accidents. Using the source terms and release factors presented, the staff calculated the radioactivity releases for each accident considered.

### 4.5.4 Radiation Doses

An analysis similar to that in Section 4.4 was used to calculate radiation doses that would be received at the fenceline of the waste facility from the releases calculated for the accidents considered. These doses are summarized in Table 4.6. The large differences in magnitude resulted from comparing accidents involving single containers with accidents involving the whole surge pool.

The total radiation doses expected from accidents in the waste facilities were calculated by multiplying the doses obtained above by the probabilities of the various accidents and by either (1) the total throughput of the waste form, taken from Appendix C, for accidents involving single containers, or (2) the estimated number of "facility years" of operation for accidents involving whole facilities. For each option, the expected doses from applicable accidents were summed to obtain a total expected dose. The results are presented in Table 4.7.

The doses calculated are all of the same magnitude. The smallest dose would occur for the full-recycle option, while the highest would occur for surface caisson storage. It is important to note that surface caisson storage is not a "closed" option, since whatever decision is made on final disposition of the spent fuel assemblies, additional processing through a waste facility would be required. This would increase the expected dose by an amount similar to that for one of the other options.

### References

1. "Reactor Safety Study: An Assessment of Accident Risks in U. S. Commercial Nuclear Power Plants," U. S. Nuclear Regulatory Commission, WASH-1400 (NUREG-75/014), October 1975.

2. "Determination of Performance Criteria for High-Level Solidified Nuclear Waste," Lawrence Livermore Laboratory, LLL NUREG-1002, April 1977.

3. "Management of Commercial High Level and Transuranic-Contaminated Radioactive Waste" (Draft), U.S. Atomic Energy Commission, WASH-1539, September 1974.

Table 4.5. Release Factors Used in Accident Analysis Calculations

| Source | Type of Radioactivity | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| | H-3, Kr-85 | | I-129 | | Volatiles | | Particulates | |
| | Building | Atmosphere | Building | Atmosphere | Building | Atmosphere | Building | Atmosphere |
| Fuel Assembly | | | | | | | | |
| Drop in pool | 0.2 | 1.0 | 0.2 | 0.01/0.5[a] | 0 | -- | 0 | -- |
| Drop in encapsulation cell | 0.2 | 1.0 | 0.2 | 0.01/0.5 | $2 \times 10^{-3}$ | $10^{-9}$/0.01 | $2 \times 10^{-4}$ | $10^{-8}$/0.01 |
| Drop during surface emplacement | 0.2 | 1.0 | 0.2 | 0.5 | 0 | -- | 0 | -- |
| Drop during deep mine emplacement | 0.2 | 1.0 | 0.2 | 0.01/0.5 | 0 | -- | 0 | -- |
| Heavy object drop onto spent assembly in pool | 1.0 | 1.0 | 1.0 | 0.01/0.5 | 0 | -- | 0 | -- |
| Aircraft impact--surface caisson | 1.0 | 1.0 | 1.0 | 1.0 | $10^{-3}$ | 1.0 | $10^{-4}$ | 1.0 |
| Spent Fuel Surge Pool | | | | | | | | |
| Loss of circulation cooling | 1.0 | 1.0 | 1.0 | 0.01/0.5 | $10^{-3}$ | $10^{-9}$/0.01 | $10^{-4}$ | $10^{-8}$/0.01 |
| Earthquake | 1.0 | 1.0 | 1.0 | 0.01/0.5 | $10^{-3}$ | 0.1 | $10^{-4}$ | 0.1 |
| Aircraft impact | 1.0 | 1.0 | 1.0 | 0.01/0.5 | $10^{-3}$ | 0.1 | $10^{-4}$ | 0.1 |
| HLSW Canister | | | | | | | | |
| Drop in overpack cell | -- | -- | -- | -- | $10^{-3}$ | $10^{-9}$/0.01 | $10^{-4}$ | $10^{-8}$/0.01 |
| HLSW Surge Pool | | | | | | | | |
| Loss of circulation cooling | -- | -- | -- | -- | $10^{-3}$ | $10^{-9}$/0.01 | $10^{-4}$ | $10^{-8}$/0.01 |
| Earthquake | -- | -- | -- | -- | $10^{-3}$ | 0.1 | $10^{-4}$ | 0.1 |
| Aircraft impact | -- | -- | -- | -- | $10^{-3}$ | 0.1 | $10^{-4}$ | 0.1 |

[a] 0.01/0.5 means a release of 0.01 when the filtration system works and a release of 0.5 when it fails.

Table 4.6. Radiation Doses from Accidents

| Accident[a] | Dose (rem) H-3 (whole body) | Kr-85 (skin) | I-129 (thyroid) | Cs (lung) | Sr-90 (bone) | α-emitters (lung) |
|---|---|---|---|---|---|---|
| SFA[a] Drop in Surge Pool | | | | | | |
| Filtration system works | $3.63 \times 10^{-6}$ | $1.15 \times 10^{-4}$ | $4.38 \times 10^{-7}$ | -- | -- | -- |
| Filtration system fails | $3.63 \times 10^{-6}$ | $1.15 \times 10^{-4}$ | $2.19 \times 10^{-5}$ | -- | -- | -- |
| Drop of Heavy Object in SFA Surge Pool | $1.82 \times 10^{-5}$ | $5.73 \times 10^{-4}$ | $2.19 \times 10^{-6}$ | -- | -- | -- |
| Loss of SFA Surge Pool Cooling | | | | | | |
| Filtration system works | $5.9 \times 10^{-2}$ | 1.87 | $7.12 \times 10^{-3}$ | $1.03 \times 10^{-7}$ | $1.62 \times 10^{-6}$ | $1.12 \times 10^{-6}$ |
| Filtration system fails | $5.9 \times 10^{-2}$ | 1.87 | $3.56 \times 10^{-1}$ | 1.03 | 1.62 | 1.12 |
| Earthquake Damage to SFA Surge Pool | $4.21 \times 10^{-1}$ | $1.34 \times 10^{+1}$ | 2.54 | $7.35 \times 10^{+1}$ | $1.16 \times 10^{+2}$ | $8.02 \times 10^{+1}$ |
| Aircraft Impact on SFA Surge Pool | $4.21 \times 10^{-1}$ | $1.34 \times 10^{+1}$ | 2.54 | $7.35 \times 10^{+1}$ | $1.16 \times 10^{+2}$ | $8.02 \times 10^{+1}$ |
| Drop of SFA in Encapsulation Cell | | | | | | |
| Filtration system works | $3.64 \times 10^{-6}$ | $1.15 \times 10^{-4}$ | $4.38 \times 10^{-7}$ | $6.32 \times 10^{-11}$ | $9.98 \times 10^{-10}$ | $6.90 \times 10^{-10}$ |
| Filtration system fails | $3.64 \times 10^{-6}$ | $1.15 \times 10^{-4}$ | $2.19 \times 10^{-5}$ | $6.32 \times 10^{-4}$ | $9.98 \times 10^{-4}$ | $6.90 \times 10^{-4}$ |
| Drop of SFA in Surface Caisson Facility | $2.60 \times 10^{-5}$ | $8.20 \times 10^{-4}$ | $1.56 \times 10^{-4}$ | -- | -- | -- |
| Aircraft Impact on SFA Caisson Storage Facility | $1.30 \times 10^{-4}$ | $4.09 \times 10^{-3}$ | $1.56 \times 10^{-3}$ | $2.26 \times 10^{-1}$ | $3.56 \times 10^{-1}$ | $2.46 \times 10^{-1}$ |
| Drop of SFA in Underground Storage Facility | | | | | | |
| Filtration system works | $3.64 \times 10^{-6}$ | $1.15 \times 10^{-4}$ | $4.38 \times 10^{-7}$ | -- | -- | -- |
| Filtration system fails | $3.64 \times 10^{-6}$ | $1.15 \times 10^{-4}$ | $2.19 \times 10^{-5}$ | -- | -- | -- |
| Loss of HLSW[a] Pool Cooling | | | | | | |
| Filtration system works | -- | -- | -- | $1.03 \times 10^{-7}$ | $1.61 \times 10^{-2}$ | $2.86 \times 10^{-7}$ |
| Filtration system fails | -- | -- | -- | 1.03 | 1.61 | $2.86 \times 10^{-1}$ |
| Earthquake Damage to HLSW Storage Pool | -- | -- | -- | $7.35 \times 10^{+1}$ | $1.15 \times 10^{+2}$ | $2.04 \times 10^{+1}$ |
| Aircraft Impact on HLSW Storage Pool | -- | -- | -- | $7.35 \times 10^{+1}$ | $1.15 \times 10^{+2}$ | $2.04 \times 10^{+1}$ |
| HLSW Canister Drop in Overpack Cell | | | | | | |
| Filtration system works | -- | -- | -- | $2.15 \times 10^{-10}$ | $3.37 \times 10^{-9}$ | $5.96 \times 10^{-10}$ |
| Filtration system fails | -- | -- | -- | $2.15 \times 10^{-3}$ | $3.37 \times 10^{-3}$ | $5.96 \times 10^{-4}$ |

[a]SFA = spent fuel assembly
HLSW = high-level solidified waste

Table 4.7. Total Radiation Doses Expected from Accidents at Waste Repositories

| Option | Expected Total Dose, rem |
|---|---|
| (1) U-only recycle, Pu stored | 0.054 |
| (2) U-only recycle, Pu disposed | 0.054 |
| (3) Full recycle | 0.045 |
| (4) No-recycle, surface storage | 0.108 |
| (5) No-recycle, deep stowaway | 0.107 |
| (6) No-recycle, deep throwaway | 0.107 |

# 5. WASTE INVENTORY*

The waste management schedule used in this report and in a computer program designed to calculate the amounts of wastes produced and the time frames for discharge and movement of the wastes to repositories is described in this section. The number of repositories that would be required and the resulting land and salt commitments (Sec. 6) were calculated for each option on the basis of the results of these computer calculations.

The program results are contained in nine computer output sections (reproduced in Appendix C). Output section 1 is for the no-recycle options; sections 2-5 are the outputs for the U-only recycle options; and sections 6-9 are the outputs for the full recycle option. The results are summarized graphically in Figures 5.1 through 5.3. In these figures, the curves for reprocessing wastes are given in terms of the amount of fuel reprocessed, not the amount of waste produced from reprocessing. For example, it is indicated in Figure 5.2 (U-only recycle) that 237,000 MT (260,000 tons) of spent fuel would have been reprocessed up to the year 2020. This does not mean that 237,000 MT of wastes would have accumulated from reprocessing.

## 5.1 NO-RECYCLE OPTIONS

By definition, the no-recycle waste schedule involves only spent PWR and BWR fuel assemblies--no reprocessing wastes. The predicted spent fuel assembly inventory is contained in section 1 of the computer output in Appendix C. The only assumptions for this option involve the PWR to BWR ratio and the nuclear reactor growth schedule.

The buildup of spent fuel discharged from power reactors and subsequently stored in repositories after a ten-year cooling period is shown graphically in Figure 5.1. The values shown have been used to determine the number of Federal repositories that would be needed to store the spent fuel assemblies (Sec. 6).

## 5.2 U-ONLY RECYCLE

Under this option, the spent fuel would be reprocessed and the recovered uranium used as fuel. The plutonium recovered would be either stored or disposed. The cumulative amount of wastes produced, the status of spent fuel assemblies, and the annual and cumulative shipments of wastes to Federal repositories are shown in sections 2 through 5 of the computer output (Appendix C). The spent fuel assembly backlog, the amount of fuel reprocessed, and the shipment of reprocessing wastes to Federal repositories are shown in Figure 5.2.

### 5.2.1 Spent Fuel Status

The spent fuel status for the uranium-only recycle options and the reprocessing schedule developed are contained in output section 2. The spent fuel assembly "status" refers to the cumulative number and amount (metric tons) of spent assemblies remaining in storage after reprocessing of the amounts specified. The number of assemblies in storage can be determined on the basis of the amount of uranium in storage and the ratios for BWR and PWR fuel assemblies per MTHM reprocessed. Examination of output section 2 indicates that by the year 2005 there would be a ten-year backlog of spent fuel awaiting reprocessing. Thus, fuel discharged after 2005 would not be reprocessed until it had been out of the reactors for at least ten years, and as a result, additional storage time for the reprocessing wastes from this fuel would not be required (see Fig. 5.2). Since reprocessing is assumed to terminate in the year 2030, the spent fuel not reprocessed by then would require permanent disposal.

*Information in this section was based on References 1-3.

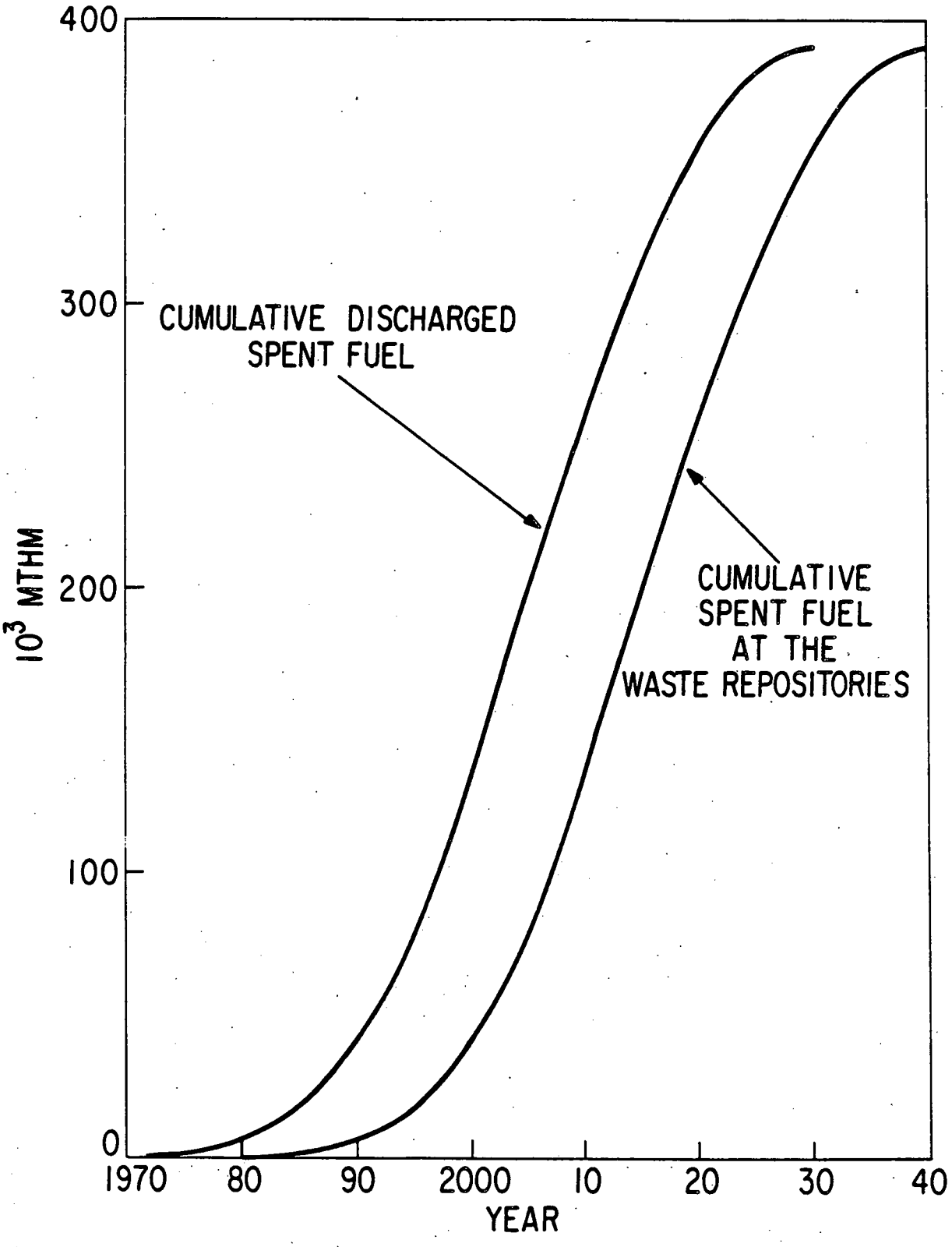


Fig. 5.1. Spent Fuel Inventory for the No-Recycle Options.

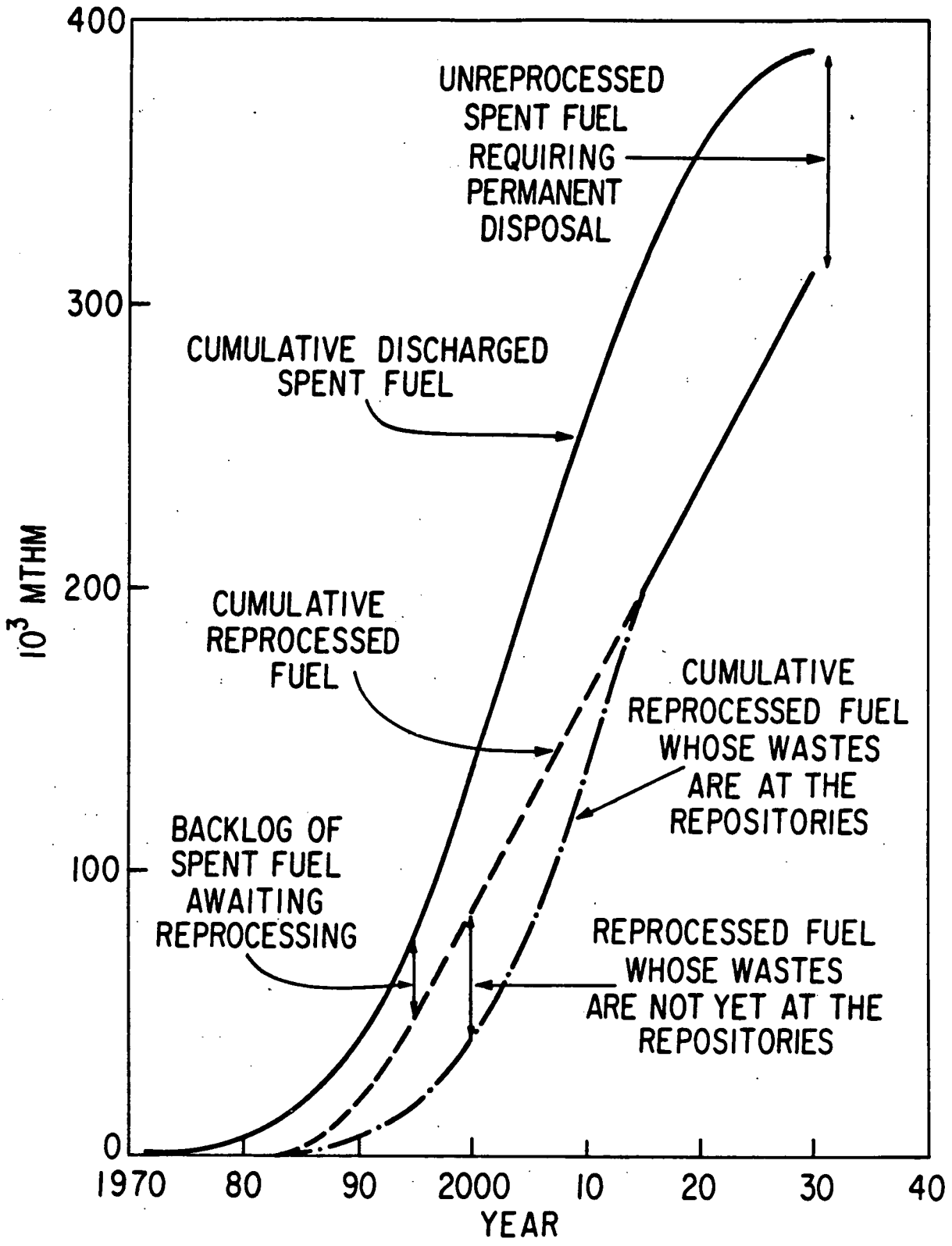


Fig. 5.2. Spent Fuel and Reprocessing Schedule for the U-Only Recycle Options.

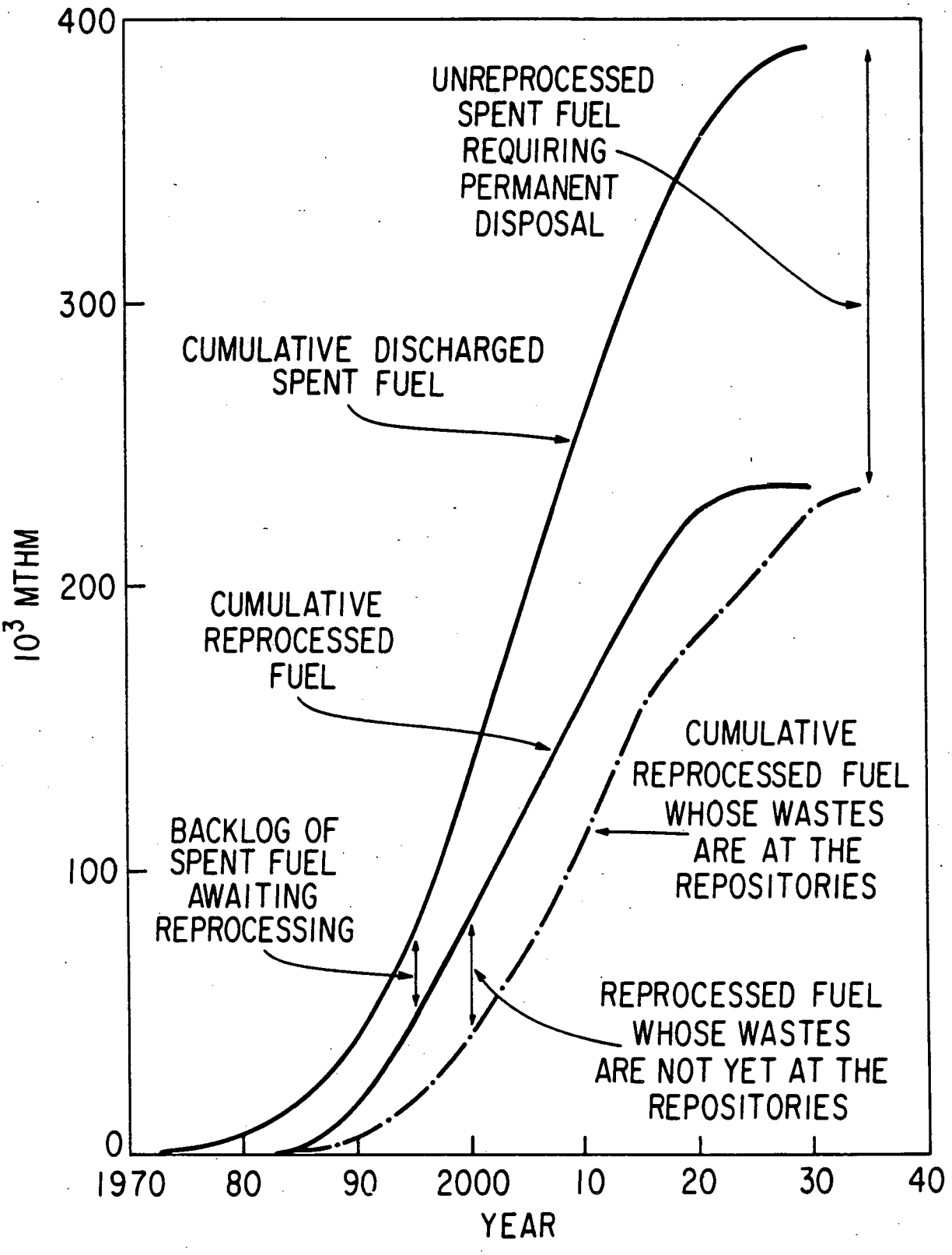


Fig. 5.3. Spent Fuel and Reprocessing Schedule for the Full Recycle Option.

### 5.2.2 Cumulative Wastes Produced by Uranium Recycle

The cumulative amounts of wastes produced during the U-only recycle options are shown in output section 3. These values are the end-of-year figures and do not contain any time-delay factors. The waste categories are:

(1) High-level waste (HLW),

(2) Hulls,

(3) Transuranic-contaminated wastes (TRU), and

(4) Plutonium.

HLW would contain 98.5% of the fission products and actinides in the spent fuel (with the exception of 100% of the tritium and noble gases, and 99.9% of the iodine and bromine) and 0.5% of the uranium and plutonium. The remaining 1.5% of the fission products and actinides would be left in the plutonium. The HLW would be processed into a calcined form and placed into canisters [0.177 $m^3$ (6.3 $ft^3$) of calcined waste per canister at 80% capacity]. This HLW would be generated at a rate of 0.0559 $m^3$/MTHM reprocessed.

The hulls, the fuel cladding, and associated fuel assembly hardware are assumed to be contaminated with 0.1% of the fuel. After being chopped and uncompacted the hulls would have a density of 1000 kg/$m^3$. Hull waste would be generated at the rate of 0.326 $m^3$/MTHM reprocessed. The hulls also would be packed into waste canisters.

There are two types of transuranic wastes (TRU)--the TRU intermediate-level wastes (TRU-ILW) and the TRU low-level wastes (TRU-LLW). The TRU-ILW require shielded handling, and the TRU-LLW do not. The TRU-ILW would be produced at a rate of 0.283 $m^3$/kg of plutonium processed, while the TRU-LLW would be produced at a rate of 0.113 $m^3$/kg of plutonium processed. The TRU wastes would all be treated uncompacted and unincinerated with a density of 1000 kg/$m^3$. The TRU-ILW would be placed in waste canisters, and the TRU-LLW would be placed in 55-gallon drums.

The plutonium waste would contain 95% plutonium (by weight) and 5% HLW. It is assumed that 10 kg (22 lb) of this spiked plutonium would be produced per MTHM reprocessed. The plutonium would be packaged 6 kg (13 lb) per container.

Also included in output section 3 is an estimation of the amount of uranium recovered. This estimation is based on the isotopic content of the spent fuel.

### 5.2.3 Waste Receiving Schedules

Output sections 4 and 5 involve the waste receiving schedule at the Federal repositories. It is assumed that the wastes would be at least ten years old before they would be accepted at a repository. This ten-year period is based on the overall out-of-reactor time for the material, not on the post-reprocessing time. The amount of wastes shipped is based upon the part of the schedule which relates to reprocessing.

Output section 4 shows the amount of wastes that would arrive at the waste repositories annually. Output section 5 presents an accounting of the cumulative amount of wastes at the repositories (this is shown in Fig. 5.2). Up until the year 2015, it would be necessary to store the reprocessing waste for an interim period until it had been out of the reactor for ten years. After 2015, this interim storage would not be required because the spent fuel would have been out of the reactor for at least ten years before being reprocessed. Output section 5 is of importance for two reasons: (1) it agrees with the results of output section 3 as to the final amounts of waste in storage, (2) it is useful in determining the schedules for construction and the capacity and number of Federal repositories required (Sec. 6).

## 5.3 FULL RECYCLE OPTION

Under this option, both the uranium and plutonium recovered from reprocessing would be used as nuclear fuel. The schedule for the full recycle option, referred to as mixed oxide (MOX) reprocessing, is contained in output sections 6 through 9. The spent fuel assembly backlog, the amount of fuel reprocessed, and the shipment of these reprocessing wastes to Federal repositories are shown graphically in Figure 5.3.

### 5.3.1 MOX Fuel Status

The projected MOX fuel status for each generation of MOX fuels is shown in output section 6. Both the generation rates of MOX fuels (MOX 1, MOX 2, and MOX 3) and the decrease in reprocessing that occurs with the decline in demand are illustrated.

The demand value (in MT) for any year is obtained from output section 6 by multiplying 26 MTHM per GWe by the "GW OK for MOX" value two years hence, at which time the reprocessed MOX would be available. However, the amount of MOX fuel actually loaded might be less than this demand value, depending upon the reprocessing capabilities. The "GW OK for MOX" value is derived from the output section 1 schedule, with the constraints for using MOX fuel given in Section 1.4.1.2. To begin the full recycle scenario, only those reactors between three and ten years old would be able to use MOX fuel once reprocessing began. Thus, in 1982 the total operating capacity (GWe) of reactors three to ten years old would be: 62.0 - 7.4 = 54.6. The value for each subsequent year is obtained by adding the new capacity from three years previous. Thus, for the year 1983, the value would be: 54.6 + 8.0 = 62.6 GWe. For calculations for the year 2000 and after, it is necessary to subtract the capacity of reactors more than 28 years old.

It should be noted that three other spent fuel waste types would occur under this option: MOX 1, MOX 2, and MOX 3. The respective cumulative totals would be 7923, 5688, and 56,148 MT. These values are readily obtained from the "totals" line of output section 6. The MOX 1 and MOX 2 wastes would be the result of an excess in MOX supply over demand as reactors shut down because of age. In addition, the total amount of unreprocessed $UO_2$ spent fuel in this option can be calculated by subtracting the total amount under the MOX reprocessing scenario (output section 6) from the total amount required. Hence, the amount of unreprocessed $UO_2$ spent fuel would be: 390,240 - 99,181 - 84,520 - 65,274 - 56,148 = 85,117 MT. The first item in this calculation is the cumulative amount of discharged $UO_2$ from output section 1. The four numbers subtracted from this amount come from output section 6 and are the amount of $UO_2$ reprocessed and the amount of MOX 1, MOX 2, and MOX 3 produced.

### 5.3.2 Full Recycle Waste Output

Output section 7 contains the schedule of waste amounts produced for the full recycle option. It is a cumulative year-end accounting of the major waste streams. The wastes considered are:

(1) High-level wastes (HLW) - from $UO_2$ and MOX fuels,

(2) Hulls,

(3) Transuranic-contaminated wastes (TRU).

The HLW is separated into two categories. The first type, HLW-$UO_2$, consists of high-level waste resulting from the reprocessing of normal uranium oxide fuels. The HLW-MOX waste results from the reprocessing of MOX fuels. The amounts of HLW produced are approximately 0.0565 $m^3$/MTHM for both HLW-$UO_2$ and HLW-MOX.

The hulls waste is generated at a rate of 0.326 $m^3$/MTHM and is chopped and uncompacted.

The TRU wastes are separated into two categories as defined above. The TRU-ILW are produced at a rate of 0.283 $m^3$/kg of plutonium processed. The TRU-LLW are produced at a rate of 0.952 $m^3$/kg of plutonium processed.

### 5.3.3 Federal Repository Waste-Receiving Schedule

Output sections 8 and 9 contain the predicted schedule for the shipment of wastes to Federal repositories. The reprocessing wastes are assumed to have been out of the reactor ten years before shipment to a repository. To simplify the inventory, it is assumed that the oldest MOX fuels would be reprocessed first. The MOX fuels would be reprocessed immediately and the wastes produced would be stored at the reprocessing facility until the ten-year total time requirement was fulfilled. As shown in output section 7, reprocessing of spent MOX fuel would begin in the year 1988, and the resulting wastes would not be received at a Federal repository until 1998. However, reprocessing wastes from $UO_2$ fuel would be at a Federal repository as early as 1982 (see Fig. 5.3).

The predictions in output sections 6 and 9 are useful in determining the number of Federal repositories required (Sec. 6).

References

1. "Final Generic Environmental Statement on the Use of Recycle Plutonium in Mixed Oxide Fuel in Light Water Cooled Reactors" (GESMO), Volumes 2 and 3, U.S. Nuclear Regulatory Commission, NUREG-0002, August 1976.

2. C. W. Kee, A. G. Croff, and J. O. Blomeke, "Updated Projections of Radioactive Wastes to be Generated by the U.S. Nuclear Power Industry," Oak Ridge National Laboratory, ORNL-TM-5427, December 1976.

3. "Alternatives for Managing Wastes from Reactors and Post-Fission Operations in the LWR Fuel Cycle," Volume 2, U.S. Energy Research and Development Administration, ERDA 76-43, May 1976.

# 6. LAND AND SALT COMMITMENTS

The amount of land and salt* committed for the storage of nuclear wastes would depend on the fuel cycle option chosen. To assess these commitments, the six options considered can be grouped into four categories:

- two no-recycle deep geologic options,
- two U-only recycle options,
- the full recycle option, and
- the no-recycle surface storage option.

The number of repositories needed and the land and salt committed for these four categories will be compared in this section.

## 6.1 DEEP GEOLOGIC STORAGE

The size of each waste repository would be about the same for all five deep geologic burial options. The small differences that would occur, such as larger mine rooms being required for the spent fuel canisters than for the reprocessing wastes canisters, are ignored in this analysis. However, because the radioactive and thermal characteristics of the waste material produced would differ under the various options considered, the burial density and amount of waste material to be handled also would vary, and thus the number of waste repositories required would depend upon the fuel cycle chosen.

Each deep geologic waste repository would be an underground excavation with a floor area of approximately 800 ha (2000 acres). Surface and subsurface activity would be strictly monitored. Surrounding the 800 ha surface zone above the deep storage area would be a restricted area of an additional 1200 ha (3000 acres). All underground activities and certain aboveground activities would be controlled within this buffer zone. Thus, the surface land committed for each waste repository would total approximately 2000 ha (5000 acres). The total land commitment for the various options can be determined by calculating the number of waste repositories required and then multiplying this value by 2000 ha per repository. The total amount of rock salt committed is taken to be that salt under the 2000 ha of surface area committed for each repository. The amount of salt not used for backfilling would be very small compared with the total amount of salt committed and is, therefore, not taken into account.

For the no-recycle options, the spent fuel assemblies would be left intact and would constitute the waste of concern. Based on information presented in Section 3, the emplacement density of spent fuel for a no-recycle $UO_2$ fuel cycle can be calculated to be 48.0 MTU/ha (19.4 MTU/acre). Based on the projected nuclear power growth estimates given in Section 1.4, storage space would be required for a total of about 390,000 MT of spent fuel that would be discharged from power reactors by the year 2030 (see Sec. 1 of App. C). Since a waste repository would have a floor area of 800 ha in which spent fuel could be stored, a 48.0 MTU/ha storage density would allow storage of about 39,000 MTU at each repository. Hence, ten repositories would be required to handle the spent fuel discharged from nuclear power reactors by the year 2030. This would result in a total of 20,000 ha (50,000 acres) of land being committed for the storage of spent fuel. Based on the stratigraphy given in Section 3, a total of $1.3 \times 10^{10}$ MT ($1.4 \times 10^{10}$ ST) of rock salt would be committed for waste storage purposes at each repository. For ten repositories, the total amount of rock salt committed would be $1.3 \times 10^{11}$ MT ($1.4 \times 10^{11}$ ST). Since the available salt reserves in the United States are estimated at $5.5 \times 10^{13}$ MT ($6.1 \times 10^{13}$ ST),[1] 0.23% would be committed to the storage of nuclear wastes.

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*Commitment of salt would be the result of using underground salt formations as the locations of waste repositories.

For the two U-only recycle options, the plutonium would be stored or disposed. The types of reprocessing wastes handled would be (1) high-level solid waste (HLSW), (2) hulls, (3) transuranic intermediate-level waste (TRU-ILW), (4) transuranic low-level waste (TRU-LLW), and (5) spiked $PuO_2$. The storage densities for the HLSW and spiked $PuO_2$ would be determined by thermal considerations (see Sec. 3). Since these two types of wastes would have different isotopic compositions, their storage densities would be different. The storage densities for the other types would be determined by mechanical considerations and are taken from Reference 2. The storage densities are shown in Table 6.1.

Table 6.1. Storage Densities of Wastes from Uranium Reprocessing

| Waste Type | Storage Density (cans/acre) |
|---|---|
| HLSW | 30.2 |
| Hulls | 748 |
| TRU-ILW | 748 |
| TRU-LLW | 6625[a] |
| $PuO_2$ | 37.1 |

[a]Drums stacked in rooms.

On the basis of the nuclear power growth projections used in this report and the storage densities shown in Table 6.1, 12 waste repositories would be required to handle the reprocessing wastes from nuclear power generation through the year 2030. Because of the limited capacity of the reprocessing plants, not all of the spent uranium would be reprocessed by the year 2030, which is the projected end of the commercial LWR nuclear power industry (see Fig. 5.2). It is assumed that this unreprocessed spent fuel would be stored in deep geologic salt formations. Two such repositories would be required, for a total of 14 waste repositories. This would result in a land commitment of about 28,000 ha (70,000 acres) and a rock salt commitment of $1.8 \times 10^{11}$ MT ($2.0 \times 10^{11}$ ST). This rock salt commitment is 0.33% of the total salt reserves in the United States.

The third category considered, full recycle of plutonium and uranium, would result in waste types similar to those for the U-only recycle case, except that the $PuO_2$ would be treated as a fuel source, not as a waste material. The composition of the HLW would be different because of the use of recycled plutonium. This would increase the amount of actinides in the fuel, which would make the high-level waste more radioactive than for the U-only recycle options. Hence, a lower storage density of the HLSW from the MOX fuel reprocessing would be required to maintain comparable heat loads. The HLSW from the recycled uranium fuel could be buried at the same density as that used for U-only recycle shown in Table 6.1. The storage densities of the waste materials from MOX fuel reprocessing are shown in Table 6.2.

Table 6.2. Storage Densities of Wastes from Mixed Oxide Fuel Reprocessing

| Waste Type | Storage Density (cans/acre) |
|---|---|
| HLSW | 11.8 |
| Hulls | 748 |
| TRU-ILW | 748 |
| TRU-LLW | 6625[a] |

[a]Drums stacked in rooms.

Seven repositories would be required to handle the reprocessing wastes with the full recycle of uranium and plutonium; however, the amount of MOX fuel obtained from the recycled uranium and plutonium would not be sufficient to fuel the operating nuclear power plants, and it thus would be necessary to augment this fuel cycle with additional uranium. The deficiency in MOX fuel in the recycling streams would be due to (1) limited reprocessing facilities, (2) the growth of the nuclear industry, and (3) the supplemental plutonium required for the 1.15 SGR MOX fuel cycle assumed. The spent fuel from this additional uranium, along with the unreprocessed spent MOX fuel (all MOX 3, plus MOX 1 and MOX 2 in excess of that needed for producing additional fuel) would have to be stored in spent fuel repositories (see Fig. 5.3). Since the spent MOX fuel would have a greater buildup of actinides because of the use of recycled plutonium as a fuel, the spent MOX fuel would produce more decay heat than would the spent uranium fuel. It therefore would be necessary that the burial concentration of the MOX fuel be lower than that used for the uranium fuel. Based on information presented in Section 3, the burial densities of spent uranium and MOX fuels are calculated to be 48.0 MTHM/ha (19.4 MTHM/acre) and 21.4 MTHM/ha (8.67 MTHM/acre), respectively. Based on these burial densities, six repositories would be needed for the disposal of the spent unreprocessed fuel, resulting in a total of 13 waste repositories. The land and salt commitments in this option would be 26,000 ha (65,000 acres) of land and $1.7 \times 10^{11}$ MT ($1.9 \times 10^{11}$ ST) of rock salt, which is 0.31% of the total available salt reserves in the United States.

## 6.2 NO-RECYCLE SURFACE STORAGE

In addition to deep geologic burial of unreprocessed spent fuel, surface storage in caissons also is considered in this report. In this method, the spent fuel assemblies would be buried in lined holes on the earth's surface. As in the deep geologic burial of spent fuel, the assemblies would be left intact. The decay heat generated by the spent fuel assemblies would be conducted to the earth's surface and dissipated to the atmosphere.

To determine the land commitment in this method of storing spent fuel, a surface waste repository is assumed to have the same areal extent as that used for deep geologic burial. That is, 800 ha (2000 acres) in each repository would be used to store spent fuel, but the total land commitment would be 2000 ha (5000 acres). Based on the GESMO projection of 507 GWe nuclear power generation by the year 2000, a total of approximately 1,230,000 spent fuel assemblies would require storage by the year 2030 (see Sec. 1 of App. C). Since the caissons would be placed 7.6 m (25 ft) apart in the repository, a total of nine repositories would be required. This would result in a total land commitment of 18,000 ha (45,000 acres).

In summary, the total land and salt commitments involved in storing the wastes from the nuclear power industry would be relatively small. Depending on the fuel cycle chosen, 9 to 14 repositories would be required to store the wastes from nuclear power generation through the year 2030 (see Table 6.3). The natural resource commitment in the storage of nuclear wastes would not be sufficiently large to preclude any option.

Table 6.3. Land and Salt Commitments for Waste Repositories

| Option | Number of Repositories | Land Commitment (acres) | | Fractional Area[a] of Hanford and Savannah River Sites[b] | | Salt Commitment (MT) | % of Total U.S. Salt Reserves |
|---|---|---|---|---|---|---|---|
| | | Burial | Total | Hanford | Savannah R. | | |
| No-recycle - deep geologic burial | 10 | 20,000 | 50,000 | 0.05 | 0.10 | $1.3 \times 10^{11}$ | 0.23% |
| U-only recycle | 14 | 28,000 | 70,000 | 0.08 | 0.15 | $1.8 \times 10^{11}$ | 0.33% |
| Full recycle | 13 | 26,000 | 65,000 | 0.07 | 0.14 | $1.7 \times 10^{11}$ | 0.31% |
| No-recycle - surface storage | 9 | 18,000 | 45,000 | 0.05 | 0.09 | --- | -- |

[a]Only the burial areas are considered in these calculations.

[b]High-level defense wastes are stored at the Hanford and Savannah River sites. The Hanford site is 570 square miles and the Savannah River site is 300 square miles.

A summary of the amounts of waste at the waste repositories in the years 2000 and 2040 is given in Tables 6.4 through 6.8. Tables 6.4 through 6.6 show the amount of land required for each type of waste material for the no-recycle (Table 6.4), U-only recycle (Table 6.5), and full recycle (Table 6.6) options in the years 2000 and 2040. It should be noted that for the year 2000, only the wastes from the fuel discharged up to 1990 would be in the repositories since there would be a ten-year cooling period from discharge to disposal. A comparison of the land commitments for the six fuel cycle options is given in Tables 6.7 and 6.8, which consist of summaries of the data from Tables 6.4 through 6.6 for the years 2000 (Table 6.7) and 2040 (Table 6.8).

Table 6.4. Types and Amounts of Nuclear Wastes at the Repositories in the Years 2000 and 2040--No-Recycle

| Waste Type | Dry Surface Retrievable Storage of Spent Fuel | | Deep, Geologic Repositing of Spent Fuel[a] | |
|---|---|---|---|---|
| | Assemblies | MT | Assemblies | MT |
| Year 2000[b] | | | | |
| Spent Fuel | | | | |
| BWR | 69,040 | 13,808 | 69,040 | 13,808 |
| PWR | 61,363 | 27,614 | 61,363 | 27,614 |
| Total | 130,403 | 41,422 | 130,403 | 41,422 |
| Repository Acres Required | | | | |
| Burial[c] | 1900 | | 2100 | |
| Total | 4750 | | 5250 | |
| Number of Repositories | 0.95 | | 1.05 | |
| Year 2040[d] | | | | |
| Spent Fuel | | | | |
| BWR | 650,404 | 130,081 | 650,404 | 130,081 |
| PWR | 578,130 | 260,159 | 578,130 | 260,159 |
| Total | 1,228,534 | 390,240 | 1,228,534 | 390,240 |
| Repository Acres Required | | | | |
| Burial | 17,600 [18,000][e] | | 20,100 [20,000] | |
| Total | 44,000 [45,000] | | 50,250 [50,000] | |
| Number of Repositories | 9 | | 10 | |

[a]Retrievable and non-retrievable modes included.

[b]Only the fuel discharged from reactors up to 1990 will reach the repositories by 2000.

[c]Since the total area of a repository is 5000 acres, with an underground burial area of 2000 acres, 2.5 total acres are required for each burial acre.

[d]The year 2040 is used as the end for repository burial since there is a 10-year delay from discharging spent fuel to ultimate disposal.

[e]Numbers in brackets correspond to the area of an integer number of repositories.

Table 6.5. Types and Amounts of Nuclear Wastes at the Repositories in the Years 2000 and 2040--U-Only Recycle

| Waste Type | Number of Canisters | Repository Acres Required | |
|---|---|---|---|
| | | Burial | Total[a] |
| Year 2000[b] | | | |
| HLSW | 13,082 | 400 | 1000 |
| Hulls | 76,292 | 100 | 250 |
| TRU-ILW | 571,021 | 800 | 2000 |
| TRU-LLW | 241,093 | 40 | 100 |
| $PuO_2$ | 69,037 | 1900 | 4750 |
| Total | | 3240 | 8100 |
| | Number of repositories required: | 1.62 | |
| Year 2040[c] | | | |
| Unreprocessed spent fuel | | | |
| BWR | 132,066 (26,413)[d] | 1,400 | 3,500 |
| PWR | 117,392 (52,827) | 2,700 | 6,750 |
| HLSW | 98,220 | 3,300 | 8,250 |
| Hulls | 572,802 | 800 | 2,000 |
| TRU-ILW | 4,287,232 | 5,700 | 14,250 |
| TRU-LLW | 1,810,132 | 300 | 750 |
| $PuO_2$ | 518,333 | 14,000 | 35,000 |
| Total | | 28,200 [28,000][e] | 70,500 [70,000] |
| | Number of Repositories Required: | 14 | |

[a] Since the total area of a repository is 5000 acres, with an underground burial area of 2000 acres, 2.5 total acres are required for each burial site.

[b] Not included are the amounts of fuel discharged but not yet reprocessed (48,732 MT) and the amounts of reprocessed fuel less than ten years out of the reactor (44,578 MT). This backlog of spent fuel will be reprocessed after the year 2000, and the reprocessing wastes will be buried when they are ten years old.

[c] The year 2040 is used as the end for repository burial since there is a ten-year delay from discharge of spent fuel to ultimate disposal.

[d] Values in parentheses are amount of fuel (metric tons).

[e] Values in brackets correspond to the area of an integer number of repositories.

Table 6.6. Types and Amounts of Nuclear Wastes at the Repositories in the Years 2000 and 2040--Full Recycle

| Waste Type | Number of Canisters | Repository Acres Required: Burial | Repository Acres Required: Total[a] |
|---|---|---|---|
| Year 2000[b] | | | |
| Spent MOX 3 fuel | 0 | 0 | 0 |
| HLSW | | | |
| From $UO_2$ reprocessing | 13,222 | 400 | 1000 |
| From MOX reprocessing | 490 | 40 | 100 |
| Hulls | | | |
| From $UO_2$ reprocessing | 76,290 | 100 | 250 |
| From MOX reprocessing | 2,827 | 4 | 10 |
| TRU-ILW | 610,430 | 800 | 2000 |
| TRU-LLW | 2,176,423 | 300 | 750 |
| Total | | 1644 | 4110 |
| Number of repositories required: | | 0.82 | |
| Year 2040[c] | | | |
| Unreprocessed spent fuel | | | |
| $UO_2$ assemblies | | | |
| BWR | 142,930 (28,586)[d] | 1,500 | 3,750 |
| PWR | 127,051 (57,173) | 2,900 | 7,250 |
| MOX assemblies[e] | | | |
| BWR | 116,265 (23,253) | 2,700 | 6,750 |
| PWR | 103,347 (46,506) | 5,400 | 13,500 |
| HLSW | | | |
| From $UO_2$ reprocessing | 31,659 | 1,000 | 2,500 |
| From MOX reprocessing | 43,471 | 3,700 | 9,250 |
| Hulls | | | |
| From $UO_2$ reprocessing | 182,670 | 200 | 500 |
| From MOX reprocessing | 250,824 | 300 | 750 |
| TRU-ILW | 4,568,734 | 6,100 | 15,250 |
| TRU-LLW | 16,289,328 | 2,500 | 6,250 |
| Total | | 26,300 [26,000][f] | 65,750 [65,000] |
| Number of repositories required: | | 13 | |

[a] Since the total area of a repository is 5000 acres, with an underground burial area of 2000 acres, 2.5 total acres are required for each burial acre.

[b] Not included are the amount of fuel discharged but not yet reprocessed (48,527 MT), the amount of spent MOX 3 fuel less than ten years out of the reactor (205 MT), and the amount of reprocessed spent fuel less than ten years out of the reactor (43,043 MT).

[c] The year 2040 is used as the end for repository burial since there is a ten-year delay from discharge of spent fuel to ultimate disposal.

[d] Values in parentheses are amount of fuel (metric tons).

[e] Includes all spent MOX 3 fuel plus the amount of unreprocessed MOX 1 and MOX 2.

[f] Values in brackets correspond to the area of an integer number of repositories.

Table 6.7. Acreages Committed for Nuclear Waste Storage Facilities in the Year 2000[a]

| Waste Type | Option | | | | | |
|---|---|---|---|---|---|---|
| | No-Recycle-Surface Storage | No-Recycle-Deep Geologic Stowaway | No-Recycle-Deep Geologic Throwaway | U-Recycle, Pu Stored | U-Recycle, Pu Disposed | Full Recycle-Deep Geologic Repositing |
| Spent Fuel | | | | | | |
| $UO_2$ | 1900 | 2100 | 2100 | 0 | 0 | 0 |
| MOX | 0 | 0 | 0 | 0 | 0 | 0 |
| HLSW | | | | | | |
| $UO_2$ | | | | 400 | 400 | 400 |
| MOX | | | | 0 | 0 | 40 |
| Hulls | | | | | | |
| $UO_2$ | | | | 100 | 100 | 100 |
| MOX | | | | 0 | 0 | 4 |
| TRU-ILW | | | | 800 | 800 | 800 |
| TRU-LLW | | | | 40 | 40 | 300 |
| $PuO_2$ | | | | 1900 | 1900 | 0 |
| Burial acres | 1900 | 2100 | 2100 | 3240 | 3240 | 1644 |
| Total acres[b] | 4750 | 5250 | 5250 | 8100 | 8100 | 4110 |
| Number of repositories required | 0.95 | 1.05 | 1.05 | 1.62 | 1.62 | 0.82 |

[a] Not included in the no-recycle options is the amount of spent fuel discharged since 1990. Not included in the recycle options are the amount of spent fuel not yet reprocessed and the amount of reprocessing wastes less than ten years out of the reactor.

[b] The total acres are calculated by multiplying the burial acres by 2.5.

Table 6.8. Acreages Committed for Nuclear Waste Storage Facilities in the Year 2040

| Waste Type | Option | | | | | |
|---|---|---|---|---|---|---|
| | No-Recycle-Surface Storage | No-Recycle-Deep Geologic Stowaway | No-Recycle-Deep Geologic Throwaway | U-Recycle, Pu Stored | U-Recycle, Pu Disposed | Full Recycle-Deep Geologic Repositing |
| Spent Fuel | | | | | | |
| $UO_2$ | 17,600 | 20,100 | 20,100 | 4,100 | 4,100 | 4,400 |
| MOX | 0 | 0 | 0 | 0 | 0 | 8,100 |
| HLSW | | | | | | |
| $UO_2$ | | | | 3,300 | 3,300 | 1,000 |
| MOX | | | | 0 | 0 | 3,700 |
| Hulls | | | | | | |
| $UO_2$ | | | | 800 | 800 | 200 |
| MOX | | | | 0 | 0 | 300 |
| TRU-ILW | | | | 5,700 | 5,700 | 6,100 |
| TRU-LLW | | | | 300 | 300 | 2,500 |
| $PuO_2$ | | | | 14,000 | 14,000 | 0 |
| Burial acres | 17,600 | 20,100 | 20,100 | 28,200 | 28,200 | 26,300 |
| Total acres[a] | 44,000 | 50,250 | 50,250 | 70,500 | 70,500 | 65,750 |
| Number of repositories required | 9 | 10 | 10 | 14 | 14 | 13 |

[a]The total acres are calculated by multiplying the burial acres by 2.5.

References

1. "Mineral Facts and Problems," Bureau of Mines Bulletin 667, 1975 Edition, U.S. Dept. of the Interior, 1975.

2. "Waste Isolation Facility Description - Bedded Salt," Office of Waste Isolation, Y/OWI/SUB-76/16506, September 1976.

# 7. ECONOMIC COSTS FOR THE MANAGEMENT OF WASTES GENERATED

Comparisons of the economic costs for waste disposal for the six fuel cycle options are presented in this section. The comparisons have been developed in a manner which allows a perspective view of the waste management costs, both capital and operating, in the years 2000 and 2040. The methods used also allow a comparison of the operating costs with the cumulated value of electricity generated in those same years. The assumptions and methodology involved are described in Section 7.4.

## 7.1 IMPLICATIONS OF NUCLEAR WASTE DISPOSAL COSTS AS RELATED TO FUEL CYCLE OPTION

The cumulated capital and operating costs in the years 2000 and 2040 for the six fuel cycle options are summarized in Table 7.1. The most expensive choices at both points in time would be the two uranium-only recycle options, followed by the full-recycle option, the two no-recycle deep geologic burial options, and the no-recycle, surface storage option. The cost estimate for the latter option does not include any expenses for the ultimate disposal of the spent fuel.* The ordering of options is the same regardless of whether only the capital costs, only the operating costs, or both categories combined are being considered. This does not hinge on how the estimates of costs were developed. The range of accuracy for all capital costs is ± 30%; for the operating costs the range is higher, on the order of 100%, all on the plus side.

Capital and operating costs are shown in Table 7.2. For the U-only recycle options, the costs would be determined by the three types of repositories required (spent fuel, $PuO_2$, and HLSW repositories). For the U-only recycle options, the disposal or storage of $PuO_2$ would make up 70% of the cost through the year 2000 and 75% of the cost through the year 2040. The cost of disposing or storing $PuO_2$ is very significant even considering the potential error in the cost estimates. This high cost would be due to the low weight per canister for $PuO_2$ and the low storage density (see Table 6.1).

For the full-recycle option, the cost for storing either $UO_2$ spent fuel or MOX spent fuel that has been recycled to the point that its isotopic composition precludes further economic use, would make up 50% of the operating cost through 2040. Again, these cost components are significant within the inherent estimation errors. Disposal of spent MOX fuel would constitute 80% of the disposal costs for the full-recycle option; this is due to the lower density of storage required for such spent fuel (150.7 cans/hectare for spent $UO_2$ fuel vs. 67.5 cans/hectare for spent MOX fuel). The differences among disposal costs for HLSW, hulls, TRU-ILW, and TRU-LLW (reprocessing wastes) between U-only recycle options and the full-recycle option would be significant. The difference among these costs would be due to the higher thermal content of MOX wastes as compared with $UO_2$ wastes.

Of the five disposal options, repositing of spent fuel would be the least expensive, and the differences in operating costs of that option compared with the others would be only high enough to be barely significant within the inherent estimating errors. The costs for the recycle options would be raised significantly through year 2040 by the need to store spent fuel, even though recycle of some of the spent fuel would reduce the amount to be stored, especially under the full recycle option.

The basic information used to develop the costs given in Tables 7.1 and 7.2 is presented later in Table 7.4.

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*Assuming that this option would be employed only as an interim means of storage pending final disposition.

Table 7.1. Cumulated Capital and Operating Costs through the Years 2000 and 2040 for Disposal of Nuclear Wastes Generated by the Fuel Cycle Options Considered (1977 dollars)

| Option | Capital Costs, $10^6$ | | Operating Costs,[a] $10^6$ | |
|---|---|---|---|---|
| | Year 2000 | Year 2040 | Year 2000 | Year 2040 |
| U-only recycle[b] | 800 | 6340 | 1550 | 18,300 |
| Full recycle | 234 | 5170 | 580 | 11,780 |
| No-recycle - surface storage | 238 | 2500 | 320 | 5,160 |
| No-recycle - deep geologic burial[c] | 312 | 3120 | 390 | 7,150 |

[a]No consideration of capital cost included.

[b]Includes cost of retrievability of $PuO_2$.

[c]Includes cost of retrievability of spent fuel.

Table 7.2. Land Requirements, Capital, and Operating Costs for the Six Fuel Cycle Options for Disposal of Nuclear Wastes through the Years 2000 and 2040

| Option | Capital Costs | | | | Operating Costs[a] | |
|---|---|---|---|---|---|---|
| | Through 2000 | | Through 2040 | | Through 2000 | Through 2040 |
| | Hectares | $10^6$ | Hectares | $10^6$ | $10^6$ | $10^6$ |
| U-Only Recycle[b] | | | | | | |
| Spent fuel - $UO_2$ | 0 | 0 | 1,700 | 640 | 0 | 1,040 |
| HLSW | 160 | 160 | 1,300 | 1200 | 470 | 3,600 |
| Hulls | 40 | | 320 | | | |
| TRU-ILW | 320 | | 2,300 | | | |
| TRU-LLW | 20 | | 120 | | | |
| $PuO_2$ | 770 | 640 | 5,700 | 4500 | 1080 | 13,680 |
| Total | 1310 | 800 | 11,440 | 6340 | 1550 | 18,300 |
| Full recycle[b] | | | | | | |
| Spent fuel - $UO_2$ | 0 | 0 | 1,700 | 690 | 0 | 1,170 |
| Spent fuel - MOX | 0 | 0 | 3,300 | 2870 | 0 | 4,560 |
| HLSW - $UO_2$ | 160 | 234 | 400 | 1610 | 580 | 6,050 |
| HLSW - MOX | 20 | | 1,500 | | | |
| Hulls | 40 | | 200 | | | |
| TRU-ILW | 320 | | 2,500 | | | |
| TRU-LLW | 120 | | 1,000 | | | |
| Total | 660 | 234 | 10,600 | 5170 | 580 | 11,780 |
| No-recycle - surface storage | 770 | 238 | 7,100 | 2500 | 320 | 5,160 |
| No-recycle - deep geologic burial (includes retrievability) | 850 | 312 | 8,100 | 3120 | 390 | 7,150 |

[a]Same area applies to operating and capital costs.

[b]Repositories for spent fuel and $PuO_2$ would be separate from one another and from the HLSW repository; the HLSW repository would also contain hulls, TRU-ILW, and TRU-LLW.

## 7.2 IMPLICATIONS OF NUCLEAR WASTE DISPOSAL COSTS AS RELATED TO THE VALUE OF POWER GENERATED THROUGH THE YEARS 2000 AND 2040

In Table 7.3, waste disposal costs are compared with the value of power generated through the years 2000 and 2040. The number of GWe-years of reactor operation obtained from the fuel is compared to the operating costs of waste disposal under the various options. The operating costs would not represent even 1% of the power value for any of the options. If the estimated operating costs were low by a factor of 2, which is the maximum error projected by the staff, the prices of power would be raised by less than 2%.

Table 7.3. Comparison of Waste Disposal Costs and Value of Power Generated through the Years 2000 and 2040

| | Year 2000 | Year 2040 |
|---|---|---|
| Cumulated gigawatt-years of electrical generation[a] | 6171 | 15,323 |
| Cumulated kWe-hours of electric energy represented by GWe-years[b] | $3.5 \times 10^{13}$ | $8.7 \times 10^{13}$ |
| Value of electric energy generated @ $0.03/kWe-hour (1977 dollars) | $\$1.0 \times 10^{12}$ | $\$2.6 \times 10^{12}$ |
| Number of GWe-years equivalent to waste disposal costs (operating costs only)[c] | | |
| U-only recycle | 10 | 110 |
| Full recycle | 4 | 60 |
| No-recycle, surface storage | 2 | 30 |
| No-recycle, deep geologic storage | 2 | 40 |

[a] GWe-year = $1 \times 10^6$ kWe-year = 1000 MWe-year.

[b] GWe-year = $1 \times 10^6$ kWe-year $\times$ 8760 hr/year $\times$ 0.65 (plant factor) = $5.7 \times 10^9$ kWe-hr.

[c] Operating costs from Table 7.1 divided by $\$1.7 \times 10^8$/GWe-year derived by ($1 \times 10^6 \times 8760 \times 0.65 \times 0.03 = \$1.7 \times 10^8$/GWe-year).

## 7.3 EFFECT OF INCLUSION OF THE COST OF CAPITAL ON TOTAL COSTS

None of the comparisons based on operating costs in the tables of this section contain any consideration of the capital costs. The capital costs are listed separately in Tables 7.2 and 7.4. Present-value calculations were not made to compare the options because the ranking of options would be the same whether capital and operating costs were considered separately or combined. An underground repository is considered to be a permanent facility, so the period over which capital and operating costs would be considered for financial purposes remains an unsettled question. Clearly, the longer a facility is considered to have a "useful" life, the more important are the operating costs relative to the capital investment. For these reasons, operating cost appears to represent the area in which comparisons among options are most likely to change. Over 100 years or more, the relative initial capital cost required to implement each option is not expected to affect the outcome of this analysis.

## 7.4 ASSUMPTIONS AND METHODOLOGY

### 7.4.1 Assumptions

1. All assumptions presented in Sections 1 and 2 of this report regarding types, amounts, and age of spent fuel, plutonium, and wastes were used in the cost estimates, except that surface facilities other than the receiving station and onsite transportation are not included in the cost estimates.

2. It was assumed for all repository and storage facilities that the waste material would arrive at the receiving facility in its proper canister and would have been tested for leaks. The costs given do not include rack or canister costs, nor the costs of putting the spent fuel or wastes into the proper container prior to repositing. These costs could be large; for example, the costs for spent fuel canisters through the year 2040 could be as high as $1.5 billion, which is about 1/5 of the facility cost, and this does not include the cost of installing and enclosing the fuel assembly within the canister.[1] It has been assumed for this analysis that these costs would be attributed to spent fuel storage rather than disposal.

Table 7.4. Typical Schedules of Repository Installations Required in Indicated Periods (used as base for development of operating costs)

| Period | Spent Fuel[a] | | | Type of Repository[b] HLSW + Hulls + ILW + LLW | | | Plutonium[c] | | |
|---|---|---|---|---|---|---|---|---|---|
| | Number of Repositories | Cost for Period,[d] $10^6 | Cumulated Cost, $10^6 | Number of Repositories | Cost for Period,[d] $10^6 | Cumulated Cost, $10^6 | Number of Repositories | Cost for Period,[e] $10^6 | Cumulated Cost, $10^6 |
| Through 2000 | 1 | 390 | -- | 1.5 | 580 | -- | 1 | 1080 | -- |
| 2000-2005 | 2 | 260 | 650 | 2 | 260 | 840 | 1 | 360 | 1,440 |
| 2005-2010 | 3 | 390 | 1040 | 3.5 | 455 | 1300 | 2 | 720 | 2,160 |
| 2010-2015 | 5 | 650 | 1690 | 4.5 | 585 | 1890 | 3 | 1080 | 3,240 |
| 2015-2020 | 6 | 780 | 2470 | 5 | 650 | 2540 | 4 | 1440 | 4,680 |
| 2020-2025 | 7 | 910 | 3380 | 6 | 780 | 3320 | 5 | 1800 | 6,480 |
| 2025-2030 | 7 | 1170 | 4550 | 7 | 910 | 4230 | 6 | 2160 | 8,640 |
| 2030-2035 | 9 | 1300 | 5850 | 7 | 910 | 5140 | 7 | 2520 | 11,160 |
| 2035-2040 | 10 | 1300 | 7150 | 7 | 910 | 6050 | 7 | 2520 | 13,680 |

[a]This schedule applies to the no-recycle, deep geologic burial options.

[b]This schedule applies to the full-recycle option.

[c]This schedule applies to the U-only recycle options.

[d]At $\$26 \times 10^6$ per repository per year.

[e]At $\$72 \times 10^6$ per repository per year.

3. The economic evaluations for the no-recycle surface storage option are based on a storage area of 130 hectares per repository, not 800 hectares as assumed elsewhere in the report. This assumption was dictated by the availability of information.

### 7.4.2 Methodology

All six options discussed in this report are included in this economic evaluation. The costs for the retrievable storage of spent fuel in the no-recycle option and the retrievable storage of plutonium for the U-only recycle option were assumed to be the same as for the respective non-retrievable modes of operation. Most of the effort (and the costs) envisioned for the facilities involved would be incurred in the building of the facility and emplacement of the spent fuel, plutonium, or wastes. Retrieving these canisters would be a reversal of receiving, inspecting, and emplacing.

The technique used to present the costs, both capital and operating, was basically the same as used in GESMO;[2] i.e., to cumulate the costs through the years 2000 and 2040. Schedules for the construction of repositories were developed to match the spent fuel and waste discharge schedules, as described in Section 5, and are shown in Table 7.4. This schedule established a time for start of operations as well so that the number of operating years for each of the repositories, as they were required to come online, could be established.

The burial area for spent fuel, reprocessing wastes, and plutonium repositories was assumed to be 800 underground hectares. For the no-recycle surface storage of spent fuel, the burial area was assumed to be 130 hectares. Total capital and operating costs are the sum of such costs for individual repositories required to handle the wastes generated. The dollar values per repository are shown in Table 7.5. Two years (2000 and 2040) were selected to depict the costs incurred. The year 2000 was selected to coincide with the final year chosen in GESMO, and the year 2040 was chosen because the reactors installed by the year 2000 would have completed their life by 2030 and all fuel discharged by these reactors would have been reposited by 2040. All dollar values are as of early 1977. The time value of money is not expressed in these numbers. Likewise, the values do not include any allowance for inflation. The amounts given are useful principally for comparing the costs of the different options. Costs are presented so that they could be escalated and discounted, if necessary. Capital costs are treated as being incurred over a relatively short period (ten years or less in the case of nuclear facilities).

Capital costs for the spent fuel and reprocessing wastes repositories (includes HLSW, hulls, TRU-ILW, and TRU-LLW) were taken from Reference 1. The error in these estimates is ± 20-30%.

Capital costs for the plutonium repository compared with that for spent fuel and reprocessing wastes repositories were developed on the basis of the area required.

There is no documentation available concerning the operating costs of the repositories. The actual operating costs might be as much as double the estimates given here. For this report, these annual operating costs were calculated on the basis of the assumptions given above and the labor requirements estimated as in Reference 1. The final cost estimates were derived by the following steps:

1. Multiplying the number of workers from Reference 1 by an hourly rate[3] for hourly employees and using $20,000 per year for salaried employees.

2. Both rates (item 1) were escalated by 35% for fringe benefits.[4]

3. The number from item 2 was multiplied by 2.2 (ratio of total operating costs, not including depreciation or taxes, to labor costs plus fringe).[4]

4. Since none of the mining costs taken from References 1 and 4 included the types of surface activity contemplated at the repositories (see Sec. 2), the mining costs determined in item 3 were doubled to account for this difference and to ensure conservatism. No specific reference is available to support this procedure. The surface facilities at a repository would be designed for an annual throughput consistent with the peak spent fuel generation rate of 12,000 MT/year (to occur in about year 2000). The underground facilities must be continually expanded to accommodate the influx of newly generated waste.

Annual operating costs were estimated for each type of underground repository. Since the number of repositories required through the year 2040 would be determined by the reactor installation schedule (see Sec. 5 and Table 7.4), the total annual cost through the years 2000 and 2040 can be determined.

Table 7.5. Size and Number of Repositories Required for Disposal of Nuclear Wastes through the Years 2000 and 2040

| Repository or Storage Type | Repository Size,[a] hectares | Options | Number of Repositories Required | | Repository Capital Cost, $10^6$ | Repository Operating Cost, $10^6$/year |
|---|---|---|---|---|---|---|
| | | | 2000 | 2040 | | |
| Underground spent fuel repository | 800 | No-recycle-deep geologic burial | 1 | 10 | 312 - $UO_2$ | 26 - $UO_2$ |
| | | Full-recycle | 0 | 6 | 709 - MOX | 57 - MOX |
| | | U-only recycle | 0 | 4 | | |
| Surface storage of spent fuel | 130 | No-recycle surface storage | 6 | 55 | 40 - 1st<br>28-subsequent | 4 |
| Underground repository for HLSW, Hulls, ILW, LLW | 800 | Full-recycle | 1 | 7 | | |
| | | U-only recycle | <1 | 5 | 234 | 26 |
| Underground repository for Plutonium | 800 | U-only recycle | 1 | 7 | 640 | 72 |

[a]Does not include area set aside for exclusion purposes; total repository size is 2.5 × underground acres used. Does include access and aisleways.

If there are continuing costs (as for inspection and monitoring) after a repository has been filled and the underground areas covered with salt (or rock), and if these costs are contemplated as continuing for hundreds of years, then an appreciable effect on unit costs (dollars per canister) can be envisioned. If, for example, such inspection and monitoring costs were $3 million per year and continued for 1000 years, the unit cost could increase as much as 50%. In the case of spent fuel, the total cost of repositing through the year 2040 would be $18/kg [($7.15 × $10^9$) ÷ (3.9 × $10^8$ kg)]. It would cost an additional $8/kg [($3.0 × $10^9$) ÷ (3.9 × $10^8$kg)] to inspect and monitor over the next 1000 years.

References

1. "Capital Costs Estimates on EIS on Management of Commercially Generated Radioactive Waste," Prepared by Bechtel, Inc., San Francisco, California, for Battelle - Pacific Northwest Laboratory (Contract B-05516-A-E), Reports 18-82, 20-*, and 20-58a.

2. "Final Generic Environmental Statement on the Use of Recycle Plutonium in Mixed Oxide Fuel in Light Water Cooled Reactors" (GESMO), U.S. Nuclear Regulatory Commission, NUREG-0002, Volume 4, August 1976.

3. NUEXCO, Inc., Monthly Publication No. 111, p. 4, 31 October 1977.

4. "Estimated Costs to Produce Copper at Kennecott Alaska," pp. 15-16, U.S. Bureau of Mines, Dept. of Interior, Circular 8602.

# APPENDIX A. WASTE TYPE CHARACTERISTICS AT TIME OF EMPLACEMENT
[grams, curies, and kilowatts per MTHM (fuel) by nuclide]*

*Data presented in this appendix are based on M. J. Bell, "ORIGEN - The ORNL Isotope Generation and Depletion Code," Oak Ridge National Laboratory, ORNL-4828, May 1973.

Table A.1. Assumed Nuclide Inventory at Time of Emplacement for HLW($UO_2$)

| Nuclide | Grams MTHM(fuel) | Curies MTHM(fuel) | kW MTHM(fuel) |
|---|---|---|---|
| SE 79 | 5.57E+00 | 3.89E-01 | 1.47E-07 |
| SR 89 | 2.15E-21 | 6.37E-17 | 2.18E-22 |
| SR 90 | 4.21E+02 | 5.95E+04 | 7.80E-02 |
| Y 90 | 1.09E-01 | 5.96E+04 | 3.51E-01 |
| Y 91 | 1.17E-18 | 2.85E-14 | 1.08E-19 |
| ZR 93 | 1.13E+03 | 2.91E+00 | 3.45E-07 |
| ZR 95 | 1.47E-16 | 3.10E-12 | 1.62E-17 |
| NB 93M | 4.76E-03 | 1.35E+00 | 4.71E-07 |
| NB 95 | 1.71E-16 | 6.72E-12 | 3.24E-17 |
| NB 95M | 1.79E-19 | 6.58E-14 | 9.16E-20 |
| TC 99 | 8.37E+02 | 1.44E+01 | 2.48E-05 |
| RU106 | 1.18E-01 | 3.97E+02 | 2.36E-05 |
| RH106 | 1.12E-07 | 3.97E+02 | 4.18E-03 |
| PD107 | 2.41E+02 | 1.15E-01 | 9.54E-09 |
| CD113M | 9.06E-02 | 2.05E+01 | 2.71E-05 |
| SN119M | 4.99E-07 | 2.19E-03 | 2.31E-09 |
| SN121M | 4.15E-01 | 1.62E+01 | 1.73E-05 |
| SN123 | 3.40E-10 | 2.89E-06 | 9.85E-12 |
| SN126 | 2.08E+01 | 5.91E-01 | 6.38E-07 |
| SB125 | 6.55E-01 | 6.95E+02 | 2.82E-03 |
| SB126 | 7.35E-06 | 5.85E-01 | 7.61E-05 |
| SB126M | 7.52E-09 | 5.91E-01 | 4.03E-06 |
| TE125M | 1.60E-02 | 2.80E+02 | 4.95E-04 |
| TE127 | 1.58E-13 | 4.16E-07 | 6.71E-13 |
| TE127M | 4.46E-11 | 4.21E-07 | 2.32E-13 |
| CS134 | 6.01E+00 | 7.32E+03 | 8.29E-02 |
| CS135 | 4.27E+02 | 3.77E-01 | 1.83E-07 |
| CS137 | 9.85E+02 | 8.57E+04 | 1.40E-01 |
| BA137M | 1.49E-04 | 8.11E+04 | 3.15E-01 |
| CE144 | 3.07E-02 | 9.82E+01 | 8.38E-05 |
| PR144 | 1.33E-06 | 9.82E+01 | 7.61E-04 |
| PM147 | 6.94E+00 | 6.44E+03 | 3.32E-03 |
| SM151 | 3.80E+01 | 1.03E+03 | 1.38E-03 |
| EU152 | 2.78E-02 | 5.46E+00 | 9.78E-05 |
| EU154 | 5.49E+01 | 7.98E+03 | 8.55E-02 |
| EU155 | 4.34E-02 | 5.54E+01 | 4.65E-05 |
| GD153 | 1.73E-07 | 6.09E-04 | 8.77E-10 |
| TB160 | 1.50E-17 | 1.69E-13 | 1.44E-18 |
| HO166M | 5.59E-04 | 1.03E-03 | 1.08E-08 |

Table A.1. HLW($UO_2$) Continued

| Nuclide | Grams MTHM(fuel) | Curies MTHM(fuel) | kW MTHM(fuel) |
|---|---|---|---|
| TL207 | 8.47E-22 | 1.61E-13 | 4.66E-19 |
| TL208 | 2.02E-13 | 5.89E-05 | 1.37E-09 |
| TL209 | 1.69E-19 | 6.92E-10 | 1.13E-14 |
| PB209 | 6.92E-15 | 3.15E-08 | 3.64E-14 |
| PB210 | 1.23E-12 | 3.96E-11 | 4.13E-18 |
| PB211 | 6.53E-21 | 1.61E-13 | 5.39E-19 |
| PB212 | 1.17E-10 | 1.64E-04 | 2.35E-10 |
| PB214 | 3.06E-17 | 1.01E-09 | 2.46E-15 |
| BI210 | 7.98E-16 | 9.90E-11 | 2.61E-16 |
| BI211 | 3.99E-22 | 1.61E-13 | 5.28E-13 |
| BI212 | 1.12E-11 | 1.64E-04 | 2.64E-09 |
| BI213 | 1.67E-15 | 3.15E-08 | 1.93E-13 |
| BI214 | 2.26E-17 | 1.01E-09 | 1.41E-14 |
| PO210 | 1.87E-14 | 3.44E-11 | 2.71E-15 |
| PO211 | <1.E-20 | 4.63E-16 | 2.14E-20 |
| PO212 | <1.E-20 | 1.05E-04 | 5.55E-09 |
| PO213 | <1.E-20 | 3.09E-08 | 1.53E-13 |
| PO214 | <1.E-20 | 1.01E-09 | 4.60E-14 |
| PO215 | <1.E-20 | 1.61E-13 | 7.06E-19 |
| PO216 | 4.70E-16 | 1.64E-04 | 6.69E-09 |
| PO218 | 3.57E-14 | 1.01E-09 | 3.66E-14 |
| AT217 | 1.94E-20 | 3.15E-08 | 1.32E-12 |
| RN219 | 1.25E-23 | 1.61E-13 | 6.52E-19 |
| RN220 | 1.73E-13 | 1.64E-04 | 6.20E-09 |
| RN222 | 6.56E-15 | 1.01E-09 | 3.29E-14 |
| FR221 | 1.77E-15 | 3.15E-08 | 1.17E-12 |
| FR223 | 6.09E-23 | 2.33E-15 | 5.47E-21 |
| RA223 | 3.14E-13 | 1.61E-13 | 5.60E-19 |
| RA224 | 1.02E-09 | 1.64E-04 | 5.60E-09 |
| RA225 | 8.06E-13 | 3.16E-08 | 2.08E-14 |
| RA226 | 1.02E-09 | 1.01E-09 | 2.87E-14 |
| RA228 | 1.03E-15 | 2.41E-13 | 1.86E-20 |
| AC225 | 5.42E-13 | 3.15E-08 | 1.08E-12 |
| AC227 | 2.29E-15 | 1.67E-13 | 8.40E-20 |
| AC228 | 1.07E-13 | 2.41E-13 | 1.25E-18 |
| TH227 | 5.09E-18 | 1.61E-13 | 5.54E-19 |
| TH228 | 1.99E-07 | 1.63E-04 | 5.35E-09 |
| TH229 | 1.49E-07 | 3.19E-08 | 9.64E-13 |
| TH230 | 2.55E-05 | 4.96E-07 | 1.40E-11 |
| TH231 | 2.99E-14 | 1.53E-08 | 1.25E-14 |
| TH232 | 5.85E-06 | 6.39E-13 | 1.55E-17 |
| TH234 | 5.72E-15 | 1.32E-11 | 4.71E-19 |
| PA231 | 3.54E-11 | 1.69E-12 | 5.15E-17 |
| PA233 | 2.90E-05 | 5.94E-01 | 8.03E-07 |
| PA234 | 6.67E-21 | 1.32E-14 | 1.20E-19 |
| PA234M | 1.93E-20 | 1.32E-11 | 6.81E-17 |

Table A.1. HLW($UO_2$) Continued

| Nuclide | Grams MTHM(fuel) | Curies MTHM(fuel) | kW MTHM(fuel) |
|---|---|---|---|
| U232 | 4.2cE-06 | 9.16E-05 | 2.94E-09 |
| U233 | 4.89E-03 | 4.63E-05 | 1.35E-03 |
| U234 | 1.08E+00 | 6.67E-03 | 1.92E-07 |
| U235 | 7.39E-03 | 1.58E-03 | 4.40E-13 |
| U236 | 2.05E+01 | 1.30E-03 | 3.53E-03 |
| U237 | 1.13E-07 | 9.24E-03 | 6.13E-09 |
| U238 | 4.01E-05 | 1.34E-11 | 3.3cE-16 |
| NP237 | 3.42E+02 | 5.94E-01 | 1.74E-05 |
| NP239 | 7.43E-05 | 1.73E+01 | 2.34E-05 |
| PU236 | <1. E-20 | 2.38E-04 | <1. E-20 |
| PU238 | 3.94E+00 | 6.66E+01 | 2.21E-03 |
| PU239 | 2.65E+01 | 1.63E+00 | 5.05E-05 |
| PU240 | 1.85E+01 | 4.08E+00 | 1.27E-04 |
| PU241 | 3.78E+00 | 3.c4E+02 | 1.c0E-05 |
| PU242 | 2.23E+00 | 8.71E-03 | 2.57E-07 |
| AM241 | 7.06E+01 | 2.42E+02 | 8.08E-03 |
| AM242 | 8.55E-06 | 6.93E+00 | 9.24E-06 |
| AM242M | 7.12E-01 | 6.93E+00 | 1.97E-05 |
| AM243 | 8.98E+01 | 1.73E+01 | 6.31E-04 |
| CM242 | 1.72E-03 | 5.70E+00 | 2.10E-04 |
| CM243 | 5.13E-02 | 2.35E+00 | 8.67E-05 |
| CM244 | 1.66E+01 | 1.34E+03 | 4.70E-02 |
| CM245 | 2.53E-01 | 4.47E-02 | 1.49E-06 |
| CM246 | 3.02E-02 | 9.32E-03 | 3.06E-07 |

Table A.2. Assumed Nuclide Inventory at Time of Emplacement for SF($UO_2$)

| Nuclide | Grams MTIIM(fuel) | Curies MTHM(fuel) | kW MTIIM(fuel) |
|---|---|---|---|
| H 3 | 4.16E-02 | 4.04E+02 | 1.44E-05 |
| SE 79 | 5.57E+00 | 3.83E-01 | 1.47E-07 |
| KR 85 | 1.53E+01 | 5.95E+03 | 9.66E-03 |
| SR 89 | 2.15E-21 | 6.07E-17 | 2.13E-22 |
| SR 90 | 4.21E+02 | 5.95E+04 | 7.80E-02 |
| Y 90 | 1.09E-01 | 5.96E+04 | 3.51E-01 |
| Y 91 | 1.17E-18 | 2.85E-14 | 1.03E-19 |
| ZR 93 | 1.13E+03 | 2.91E+00 | 3.45E-07 |
| ZR 95 | 1.47E-16 | 3.10E-12 | 1.62E-17 |
| NB 93M | 4.76E-03 | 1.35E+00 | 4.71E-07 |
| NB 95 | 1.71E-16 | 6.72E-12 | 3.24E-17 |
| NB 95M | 1.73E-19 | 6.53E-14 | 9.16E-20 |
| TC 99 | 8.37E+02 | 1.44E+01 | 2.48E-05 |
| RU106 | 1.18E-01 | 3.97E+02 | 2.35E-05 |
| RH106 | 1.12E-07 | 3.97E+02 | 4.13E-03 |
| PD107 | 2.41E+02 | 1.15E-01 | 9.54E-09 |
| CD113M | 9.06E-02 | 2.05E+01 | 2.71E-05 |
| SN119M | 4.99E-07 | 2.19E-03 | 2.31E-09 |
| SN121M | 4.15E-01 | 1.62E+01 | 1.70E-05 |
| SN123 | 3.40E-10 | 2.89E-06 | 9.85E-12 |
| SN126 | 2.08E+01 | 5.91E-01 | 6.33E-07 |
| SB125 | 6.55E-01 | 6.95E+02 | 2.82E-03 |
| SB126 | 7.05E-05 | 5.85E-01 | 7.61E-06 |
| SB126M | 7.52E-09 | 5.91E-01 | 4.00E-06 |
| TE125M | 1.60E-02 | 2.88E+02 | 4.95E-04 |
| TE127 | 1.58E-13 | 4.16E-07 | 8.71E-13 |
| TE127M | 4.46E-11 | 4.21E-07 | 2.32E-13 |
| I129 | 2.27E+02 | 3.71E-02 | 2.44E-08 |
| I131 | 1.81-136 | 2.25-131 | 7.34-137 |
| XE131M | 1.70E-94 | 1.43E-89 | 2.79E-95 |
| XE133 | 2.10-208 | 3.92-203 | 5.13-209 |
| CS134 | 6.01E+00 | 7.82E+03 | 8.24E-02 |
| CS135 | 4.27E+02 | 3.77E-01 | 1.83E-07 |
| CS137 | 9.85E+02 | 8.57E+04 | 1.40E-01 |
| BA137M | 1.49E-04 | 8.01E+04 | 3.15E-01 |
| CE144 | 3.07E-02 | 9.82E+01 | 8.33E-05 |
| PR144 | 1.30E-06 | 9.82E+01 | 7.61E-04 |
| PM147 | 6.94E+00 | 6.44E+03 | 3.32E-03 |
| SM151 | 3.80E+01 | 1.03E+03 | 1.80E-03 |
| EU152 | 2.78E-02 | 5.46E+00 | 9.78E-05 |
| EU154 | 5.49E+01 | 7.98E+03 | 6.55E-02 |
| EU155 | 4.34E-02 | 5.54E+01 | 4.66E-05 |
| GD153 | 1.73E-07 | 6.09E-04 | 8.77E-10 |
| TB160 | 1.50E-17 | 1.69E-13 | 1.44E-18 |
| HO166M | 5.59E-04 | 1.00E-03 | 1.03E-08 |

Table A.2. SF($UO_2$) Continued

| Nuclide | Grams MTHM(fuel) | Curies MTHM(fuel) | kW MTHM(fuel) |
|---|---|---|---|
| TL207 | 1.69E-19 | 3.21E-11 | 9.71E-17 |
| TL208 | 1.99E-11 | 5.81E-03 | 1.35E-07 |
| TL209 | 1.69E-19 | 6.92E-10 | 1.13E-14 |
| PB209 | 6.92E-15 | 3.15E-08 | 3.64E-14 |
| PB210 | 2.17E-10 | 1.76E-08 | 7.32E-15 |
| PB211 | 1.30E-18 | 3.22E-11 | 1.06E-16 |
| PB212 | 1.16E-09 | 1.61E-02 | 2.32E-06 |
| PB214 | 5.31E-15 | 1.74E-07 | 4.24E-13 |
| BI210 | 1.41E-13 | 1.75E-08 | 4.62E-14 |
| BI211 | 7.77E-20 | 3.22E-11 | 1.25E-15 |
| BI212 | 1.10E-09 | 1.61E-02 | 2.80E-07 |
| BI213 | 1.67E-15 | 3.15E-08 | 1.93E-13 |
| BI214 | 3.90E-15 | 1.74E-07 | 2.43E-12 |
| PO210 | 3.33E-12 | 1.50E-08 | 4.81E-13 |
| PO211 | <1.E-20 | 9.66E-14 | 4.27E-18 |
| PO212 | 5.62E-20 | 1.03E-02 | 5.48E-07 |
| PO213 | <1.E-20 | 3.08E-09 | 1.53E-13 |
| PO214 | 6.60E-22 | 1.74E-07 | 7.94E-12 |
| PO215 | <1.E-20 | 3.22E-11 | 1.41E-15 |
| PO216 | 4.63E-14 | 1.61E-02 | 6.61E-07 |
| PO218 | 6.16E-15 | 1.74E-07 | 6.31E-12 |
| AT217 | 1.94E-20 | 3.15E-08 | 1.32E-12 |
| RN219 | 2.50E-21 | 3.22E-11 | 1.30E-15 |
| RN220 | 1.76E-11 | 1.61E-02 | 6.12E-07 |
| RN222 | 1.13E-12 | 1.74E-07 | 5.68E-12 |
| FR221 | 1.77E-16 | 3.15E-08 | 1.17E-12 |
| FR223 | 1.22E-20 | 4.66E-13 | 1.09E-18 |
| RA223 | 6.27E-15 | 3.22E-11 | 1.12E-15 |
| RA224 | 1.01E-07 | 1.61E-02 | 5.52E-07 |
| RA225 | 8.06E-13 | 3.16E-08 | 2.08E-14 |
| RA226 | 1.77E-07 | 1.75E-07 | 4.94E-12 |
| RA228 | 2.06E-13 | 4.82E-11 | 3.71E-18 |
| AC225 | 5.42E-13 | 3.15E-08 | 1.08E-12 |
| AC227 | 4.57E-13 | 3.33E-11 | 1.68E-17 |
| AC228 | 2.15E-17 | 4.82E-11 | 2.56E-16 |
| TH227 | 1.02E-15 | 3.22E-11 | 1.11E-15 |
| TH228 | 1.97E-05 | 1.62E-02 | 5.29E-07 |
| TH229 | 1.49E-07 | 3.19E-08 | 9.64E-13 |
| TH230 | 4.17E-03 | 3.10E-05 | 2.29E-09 |
| TH231 | 5.96E-12 | 3.16E-06 | 2.49E-12 |
| TH232 | 1.17E-03 | 1.28E-10 | 3.09E-15 |
| TH234 | 1.36E-05 | 3.14E-01 | 1.12E-07 |
| PA231 | 7.07E-03 | 3.37E-10 | 1.03E-14 |
| PA233 | 2.91E-05 | 5.96E-01 | 8.06E-07 |
| PA234 | 1.52E-10 | 3.14E-04 | 2.65E-09 |
| PA234M | 4.57E-10 | 3.14E-01 | 1.62E-06 |
| U232 | 2.56E-04 | 1.83E-02 | 5.38E-07 |
| U233 | 4.86E-03 | 4.63E-05 | 1.35E-09 |
| U234 | 1.52E+02 | 9.42E-01 | 2.71E-05 |
| U235 | 1.46E+00 | 3.16E-06 | 6.78E-11 |

Table A.2. SF($UO_2$) Continued

| Nuclide | Grams MTHM(fuel) | Curies MTHM(fuel) | kW MTHM(fuel) |
|---|---|---|---|
| U236 | 4.10E+03 | 2.60E-01 | 7.05E-06 |
| U237 | 2.27E-05 | 1.85E+00 | 1.23E-06 |
| U238 | 9.42E+05 | 3.14E-01 | 7.94E-06 |
| NP237 | 8.46E+02 | 5.97E-01 | 1.75E-05 |
| NP239 | 7.43E-05 | 1.73E+01 | 2.34E-05 |
| PU236 | 8.95E-05 | 4.76E-02 | <1.E-20 |
| PU238 | 3.12E+01 | 5.27E+02 | 1.74E-02 |
| PU239 | 5.2eE+03 | 3.24E+00 | 1.01E-02 |
| PU240 | 2.19E+03 | 4.03E+02 | 1.50E-02 |
| PU241 | 7.59E+02 | 7.71E+04 | 3.20E-03 |
| PU242 | 4.46E+02 | 1.74E+00 | 5.14E-05 |
| AM241 | 5.25E+02 | 1.80E+03 | 6.01E-02 |
| AM242 | 8.55E-06 | 6.93E+00 | 9.24E-06 |
| AM242M | 7.12E-01 | 6.93E+00 | 1.97E-06 |
| AM243 | 8.98E+01 | 1.73E+01 | 6.31E-04 |
| CM242 | 1.72E-03 | 5.70E+00 | 2.10E-04 |
| CM243 | 5.13E-02 | 2.36E+00 | 8.67E-05 |
| CM244 | 1.66E+01 | 1.34E+03 | 4.70E-02 |
| CM245 | 2.53E-01 | 4.47E-02 | 1.40E-05 |
| CM246 | 3.02E-02 | 9.32E-03 | 3.06E-07 |

Table A.3. Assumed Nuclide Inventory at Time of Emplacement for SPK Pu

| Nuclide | Grams MTHM(fuel) | Curies MTHM(fuel) | kW MTHM(fuel) |
|---|---|---|---|
| SE 79 | 8.14E-02 | 5.67E-03 | 2.15E-09 |
| SR 89 | 3.13E-23 | 8.84E-19 | 3.18E-24 |
| SR 90 | 6.13E+00 | 8.67E+02 | 1.14E-03 |
| Y 90 | 1.59E-03 | 8.69E+02 | 5.11E-03 |
| Y 91 | 1.70E-22 | 4.15E-16 | 1.58E-21 |
| ZR 93 | 1.66E+01 | 4.25E-02 | 5.04E-09 |
| ZR 95 | 2.14E-18 | 4.52E-14 | 2.37E-19 |
| NB 93M | 6.95E-05 | 1.97E-02 | 6.83E-09 |
| NB 95 | 2.50E-18 | 9.81E-14 | 4.72E-19 |
| NB 95M | 2.62E-21 | 9.60E-16 | 1.34E-21 |
| TC 99 | 1.22E+01 | 2.10E-01 | 3.61E-07 |
| RU106 | 1.73E-03 | 5.80E+00 | 3.44E-07 |
| RH106 | 1.64E-09 | 5.80E+00 | 6.10E-05 |
| PD107 | 3.52E+00 | 1.68E-03 | 1.39E-10 |
| CD113M | 1.32E-03 | 2.99E-01 | 3.96E-07 |
| SN119M | 7.28E-09 | 3.20E-05 | 3.37E-11 |
| SN121M | 6.05E-05 | 2.35E-03 | 2.47E-09 |
| SN123 | 4.95E-12 | 4.21E-08 | 1.44E-13 |
| SN126 | 3.04E-01 | 8.63E-03 | 9.31E-09 |
| SB125 | 9.56E-03 | 1.01E+01 | 4.11E-05 |
| SB126 | 1.03E-07 | 8.54E-03 | 1.11E-07 |
| SB126M | 1.10E-10 | 8.63E-03 | 5.84E-08 |
| TE125M | 2.33E-04 | 4.20E+00 | 7.22E-06 |
| TE127 | 2.31E-15 | 6.03E-09 | 9.80E-15 |
| TE127M | 6.51E-13 | 6.15E-09 | 3.39E-15 |
| CS134 | 8.78E-02 | 1.14E+02 | 1.21E-03 |
| CS135 | 6.23E+00 | 5.50E-03 | 2.67E-09 |
| CS137 | 1.44E+01 | 1.25E+03 | 2.05E-03 |
| BA137M | 2.18E-06 | 1.17E+03 | 4.61E-03 |
| CE144 | 4.50E-04 | 1.44E+00 | 1.18E-06 |
| PR144 | 1.93E-08 | 1.44E+00 | 1.11E-05 |
| PM147 | 1.02E-01 | 9.44E+01 | 4.87E-05 |
| SM151 | 5.56E-01 | 1.51E+01 | 2.64E-05 |
| EU152 | 4.06E-04 | 7.97E-02 | 1.43E-06 |
| EU154 | 8.34E-01 | 1.17E+02 | 9.58E-04 |
| EU155 | 6.34E-04 | 8.08E-01 | 6.80E-07 |
| GD153 | 2.52E-09 | 8.89E-06 | 1.23E-11 |
| TB160 | 2.19E-19 | 2.48E-15 | 2.10E-20 |
| HO166M | 8.20E-06 | 1.47E-05 | 1.58E-10 |

Table A.3. SPK Pu Continued

| Nuclide | Grams MTHM(fuel) | Curies MTHM(fuel) | kW MTHM(fuel) |
|---|---|---|---|
| TL207 | 3.70E-19 | 7.03E-11 | 2.12E-16 |
| TL208 | <1.E-20 | 4.41E-07 | 1.03E-11 |
| TL209 | 1.02E-20 | 4.18E-12 | 6.83E-17 |
| PB209 | 4.18E-17 | 1.90E-10 | 2.20E-16 |
| PB210 | 9.04E-12 | 7.33E-10 | 3.04E-17 |
| PB211 | 2.85E-18 | 7.05E-11 | 2.36E-16 |
| PB212 | 6.77E-13 | 1.22E-06 | 1.76E-12 |
| PB214 | 2.88E-16 | 9.44E-09 | 2.30E-14 |
| BI210 | 5.86E-15 | 7.27E-10 | 1.91E-15 |
| BI211 | 1.70E-19 | 7.05E-11 | 2.75E-15 |
| BI212 | 8.36E-14 | 1.22E-06 | 2.12E-11 |
| BI213 | 1.01E-17 | 1.90E-10 | 1.17E-15 |
| BI214 | 2.12E-16 | 9.44E-09 | 1.32E-13 |
| PO210 | 1.32E-13 | 5.94E-10 | 1.90E-14 |
| PO211 | <1.E-20 | 2.11E-13 | 9.34E-13 |
| PO212 | <1.E-20 | 7.63E-07 | 4.15E-11 |
| PO213 | <1.E-20 | 1.86E-10 | 9.23E-15 |
| PO214 | 3.58E-23 | 9.44E-09 | 4.30E-13 |
| PO215 | <1.E-20 | 7.05E-11 | 3.09E-15 |
| PO216 | <1.E-20 | 1.22E-06 | 5.00E-11 |
| PO218 | 3.34E-17 | 9.45E-09 | 3.42E-13 |
| AT217 | 1.17E-22 | 1.90E-10 | 7.96E-15 |
| RN219 | 5.47E-21 | 7.05E-11 | 2.85E-15 |
| RN220 | <1.E-20 | 1.22E-06 | 4.64E-11 |
| RN222 | 8.13E-14 | 9.45E-09 | 3.08E-13 |
| FR221 | 1.07E-18 | 1.90E-10 | 7.07E-15 |
| FR223 | 2.64E-20 | 1.01E-12 | 2.37E-14 |
| RA223 | 1.37E-15 | 7.05E-11 | 2.45E-15 |
| RA224 | 7.63E-12 | 1.22E-06 | 4.16E-11 |
| RA225 | 4.88E-15 | 1.32E-10 | 1.26E-15 |
| RA226 | 9.59E-09 | 9.49E-09 | 2.69E-13 |
| RA228 | 6.69E-17 | 1.57E-14 | 1.21E-21 |
| AC225 | 3.27E-15 | 1.90E-10 | 6.52E-15 |
| AC227 | 9.93E-13 | 7.24E-11 | 5.65E-17 |
| AC228 | 6.99E-21 | 1.67E-14 | 8.31E-20 |
| TH227 | 2.22E-15 | 7.02E-11 | 2.42E-15 |
| TH228 | 1.48E-09 | 1.22E-06 | 3.98E-11 |
| TH229 | 9.06E-10 | 1.94E-10 | 5.36E-15 |
| TH230 | 3.36E-04 | 6.53E-06 | 1.85E-10 |
| TH231 | 8.27E-12 | 4.39E-06 | 3.46E-12 |
| TH232 | 4.67E-07 | 5.10E-14 | 1.23E-13 |
| TH234 | 1.01E-03 | 2.35E-05 | 8.36E-12 |
| PA231 | 1.27E-03 | 6.05E-10 | 1.85E-14 |
| PA233 | 5.59E-07 | 1.14E-02 | 1.54E-09 |
| PA234 | 1.18E-14 | 2.35E-03 | 2.14E-13 |
| PA234M | 3.42E-14 | 2.35E-05 | 1.21E-10 |
| U233 | 4.49E-05 | 4.25E-07 | 1.24E-11 |
| U234 | 2.40E+01 | 1.49E-01 | 4.29E-06 |
| U235 | 2.05E+00 | 4.33E-06 | 1.22E-10 |
| U236 | 2.72E+00 | 1.73E-04 | 4.66E-09 |

Table A.3. SPK Pu Continued

| Nuclide | Grams MTHM(fuel) | Curies MTHM(fuel) | kW MTHM(fuel) |
|---|---|---|---|
| U237 | 1.30E-05 | 1.55E+00 | 1.03E-06 |
| U238 | 7.05E+01 | 2.35E-05 | 5.94E-10 |
| NP237 | 1.63E+01 | 1.15E-02 | 3.37E-07 |
| NP239 | 1.16E-06 | 2.69E-01 | 3.64E-07 |
| PU238 | 3.02E+02 | 5.09E+03 | 1.69E-01 |
| PU239 | 5.17E+03 | 3.17E+02 | 9.86E-03 |
| PU240 | 2.15E+03 | 4.74E+02 | 1.48E-02 |
| PU241 | 6.34E+02 | 6.45E+04 | 2.58E-03 |
| PU242 | 3.50E+02 | 1.36E+00 | 4.03E-05 |
| AM241 | 3.83E+02 | 1.31E+03 | 4.38E-02 |
| AM242 | 1.61E-07 | 1.30E-01 | 1.73E-07 |
| AM242M | 1.34E-02 | 1.30E-01 | 3.70E-06 |
| AM243 | 1.40E+00 | 2.69E-01 | 9.82E-06 |
| CM242 | 3.23E-05 | 1.07E-01 | 3.95E-05 |
| CM244 | 3.05E-01 | 2.47E+01 | 8.64E-04 |
| CM246 | 3.27E-03 | 1.01E-03 | 3.31E-08 |

Table A.4. Assumed Nuclide Inventory at Time of Emplacement for HLW(MOX)

| Nuclide | Grams MTHM(fuel) | Curies MTHM(fuel) | kW MTHM(fuel) |
|---|---|---|---|
| SE 79 | 5.57E+00 | 3.89E-01 | 1.47E-07 |
| SR 89 | 2.15E-21 | 6.07E-17 | 2.18E-22 |
| SR 90 | 4.21E+02 | 5.95E+04 | 7.80E-02 |
| Y 90 | 1.09E-01 | 5.96E+04 | 3.51E-01 |
| Y 91 | 1.17E-18 | 2.85E-14 | 1.08E-19 |
| ZR 93 | 1.13E+03 | 2.91E+00 | 3.45E-07 |
| ZR 95 | 1.47E-16 | 3.10E-12 | 1.62E-17 |
| NB 93M | 4.76E-03 | 1.35E+00 | 4.71E-07 |
| NB 95 | 1.71E-16 | 6.72E-12 | 3.24E-17 |
| NB 95M | 1.79E-19 | 6.53E-14 | 9.16E-20 |
| TC 99 | 8.37E+02 | 1.44E+01 | 2.48E-05 |
| RU106 | 1.18E-01 | 3.97E+02 | 2.36E-05 |
| RH106 | 1.12E-07 | 3.97E+02 | 4.18E-03 |
| PD107 | 2.41E+02 | 1.15E-01 | 9.54E-09 |
| CD113M | 9.06E-02 | 2.85E+01 | 2.71E-05 |
| SN119M | 4.99E-07 | 2.19E-03 | 2.31E-09 |
| SN121M | 4.15E-01 | 1.62E+01 | 1.70E-05 |
| SN123 | 3.40E-10 | 2.89E-06 | 9.85E-12 |
| SN126 | 2.08E+01 | 5.91E-01 | 6.38E-07 |
| SB125 | 6.55E-01 | 6.95E+02 | 2.82E-03 |
| SB126 | 7.05E-06 | 5.85E-01 | 7.61E-05 |
| SB126M | 7.52E-09 | 5.91E-01 | 8.00E-06 |
| TE125M | 1.60E-02 | 2.83E+02 | 4.95E-04 |
| TE127 | 1.58E-13 | 4.16E-07 | 6.71E-13 |
| TE127M | 4.46E-11 | 4.21E-07 | 2.32E-13 |
| CS134 | 6.31E+00 | 7.32E+03 | 8.28E-02 |
| CS135 | 4.27E+02 | 3.77E-01 | 1.83E-07 |
| CS137 | 9.85E+02 | 8.57E+04 | 1.40E-01 |
| BA137M | 1.49E-04 | 8.01E+04 | 3.15E-01 |
| CE144 | 3.07E-02 | 9.82E+01 | 8.03E-05 |
| PR144 | 1.30E-06 | 9.82E+01 | 7.61E-04 |
| PM147 | 6.94E+00 | 6.44E+03 | 3.32E-03 |
| SM151 | 3.80E+01 | 1.03E+03 | 1.80E-03 |
| EU152 | 2.78E-02 | 5.46E+00 | 9.78E-05 |
| EU154 | 5.49E+01 | 7.98E+03 | 5.55E-02 |
| EU155 | 4.34E-02 | 5.54E+01 | 4.65E-05 |
| GD153 | 1.73E-07 | 6.09E-04 | 8.77E-13 |
| TB160 | 1.50E-17 | 1.69E-13 | 1.44E-18 |
| HO166M | 5.59E-04 | 1.00E-03 | 1.08E-08 |

Table A.4. HLW(MOX) Continued

| Nuclide | Grams MTHM(fuel) | Curies MTHM(fuel) | kW MTHM(fuel) |
|---|---|---|---|
| TL207 | 1.06E-17 | 2.01E-09 | 6.09E-15 |
| TL208 | <1.E-20 | 3.00E-05 | 6.99E-10 |
| TL209 | 4.25E-19 | 1.74E-10 | 2.84E-15 |
| PB209 | 1.74E-15 | 7.90E-09 | 9.13E-15 |
| PB210 | 2.69E-10 | 2.19E-08 | 9.07E-16 |
| PB211 | 6.16E-17 | 2.02E-09 | 6.75E-15 |
| PB212 | 5.97E-11 | 8.33E-05 | 1.20E-10 |
| PB214 | 3.68E-15 | 1.21E-07 | 2.94E-13 |
| BI210 | 1.76E-13 | 2.18E-08 | 5.73E-14 |
| BI211 | 4.67E-18 | 2.02E-09 | 7.87E-14 |
| BI212 | 5.69E-12 | 8.33E-05 | 1.45E-09 |
| BI213 | 4.20E-16 | 7.90E-09 | 4.86E-14 |
| BI214 | 2.70E-15 | 1.21E-07 | 1.68E-12 |
| PO210 | 4.49E-12 | 2.01E-08 | 6.44E-13 |
| PO211 | <1.E-20 | 6.06E-12 | 2.68E-16 |
| PO212 | <1.E-20 | 5.33E-05 | 2.83E-09 |
| PO213 | <1.E-20 | 7.75E-09 | 3.84E-13 |
| PO214 | <1.E-20 | 1.21E-07 | 5.49E-12 |
| PO215 | <1.E-20 | 2.02E-09 | 8.85E-14 |
| PO216 | <1.E-20 | 8.33E-05 | 3.41E-09 |
| PO218 | 4.26E-16 | 1.21E-07 | 4.37E-12 |
| AT217 | 4.86E-21 | 7.90E-09 | 3.31E-13 |
| RN219 | 1.97E-19 | 2.02E-09 | 8.18E-14 |
| RN220 | <1.E-20 | 8.33E-05 | 3.16E-09 |
| RN222 | 7.83E-13 | 1.21E-07 | 3.93E-12 |
| FR221 | 4.46E-17 | 7.90E-09 | 2.94E-13 |
| FR223 | 7.54E-19 | 2.89E-11 | 6.77E-17 |
| RA223 | 3.93E-14 | 2.02E-09 | 7.02E-14 |
| RA224 | 5.20E-10 | 8.33E-05 | 2.85E-09 |
| RA225 | 2.03E-13 | 7.96E-09 | 5.24E-15 |
| RA226 | 1.22E-07 | 1.21E-07 | 3.42E-12 |
| RA228 | 1.18E-15 | 2.77E-13 | 2.14E-20 |
| AC225 | 1.36E-13 | 7.90E-09 | 2.71E-13 |
| AC227 | 2.83E-11 | 2.07E-09 | 1.04E-15 |
| AC228 | 1.24E-19 | 2.77E-13 | 1.47E-18 |
| TH227 | 6.35E-14 | 2.01E-09 | 6.91E-14 |
| TH228 | 1.01E-07 | 8.28E-05 | 2.71E-09 |
| TH229 | 3.77E-08 | 8.05E-09 | 2.44E-13 |
| TH230 | 1.19E-03 | 2.31E-05 | 6.53E-10 |
| TH231 | 1.27E-10 | 6.71E-05 | 5.29E-11 |
| TH232 | 6.74E-06 | 7.37E-13 | 1.78E-17 |
| TH234 | 6.71E-08 | 1.55E-03 | 5.53E-10 |
| PA231 | 3.00E-07 | 1.43E-08 | 4.36E-13 |
| PA233 | 1.92E-05 | 3.94E-01 | 5.32E-07 |
| PA234 | 7.83E-13 | 1.55E-06 | 1.41E-11 |
| PA234M | 2.26E-12 | 1.55E-03 | 7.99E-09 |
| U233 | 1.78E-03 | 1.69E-05 | 4.92E-10 |
| U234 | 4.57E+00 | 2.83E-02 | 8.14E-07 |
| U235 | 3.13E+01 | 6.71E-05 | 1.86E-09 |
| U236 | 2.37E+01 | 1.50E-03 | 4.08E-08 |

Table A.4. HLW(MOX) Continued

| Nuclide | Grams MTHM(fuel) | Curies MTHM(fuel) | kW MTHM(fuel) |
|---|---|---|---|
| U237 | 2.71E-07 | 2.22E-02 | 1.47E-08 |
| U236 | 4.66E+03 | 1.55E-03 | 3.93E-08 |
| NP237 | 5.58E+02 | 3.94E-01 | 1.16E-05 |
| NP239 | 9.19E-04 | 2.14E+02 | 2.89E-04 |
| PU238 | 5.32E+01 | 8.98E+02 | 2.98E-02 |
| PU239 | 1.15E+02 | 7.04E+00 | 2.19E-04 |
| PU240 | 2.14E+02 | 4.71E+01 | 1.47E-03 |
| PU241 | 9.08E+00 | 9.23E+02 | 3.63E-05 |
| PU242 | 1.02E+01 | 3.99E-02 | 1.13E-06 |
| AM241 | 1.53E+02 | 5.26E+02 | 1.75E-02 |
| AM242 | 1.09E-04 | 8.83E+01 | 1.18E-04 |
| AM242M | 9.08E+00 | 8.83E+01 | 2.51E-05 |
| AM243 | 1.11E+03 | 2.14E+02 | 7.80E-03 |
| CM242 | 2.19E-02 | 7.27E+01 | 2.68E-03 |
| CM244 | 4.16E+02 | 3.37E+04 | 1.18E+00 |
| CM245 | 6.61E+01 | 1.17E+01 | 3.67E-04 |
| CM246 | 6.57E+00 | 2.03E+00 | 6.66E-05 |

Table A.5. Assumed Nuclide Inventory at Time of Emplacement for SF(MOX)

| Nuclide | Grams MTIHM(fuel) | Curies MTIHM(fuel) | kW MTIHM(fuel) |
|---|---|---|---|
| H 3 | 4.16E-02 | 4.04E+02 | 1.44E-05 |
| SE 79 | 5.57E+00 | 3.83E-01 | 1.47E-07 |
| KR 85 | 1.53E+01 | 5.95E+03 | 9.66E-03 |
| SR 89 | 2.15E-21 | 6.07E-17 | 2.18E-22 |
| SR 90 | 4.21E+02 | 5.95E+04 | 7.80E-02 |
| Y 90 | 1.09E-01 | 5.96E+04 | 3.51E-01 |
| Y 91 | 1.17E-19 | 2.85E-14 | 1.03E-19 |
| ZR 93 | 1.13E+03 | 2.91E+00 | 3.45E-07 |
| ZR 95 | 1.47E-16 | 3.10E-12 | 1.62E-17 |
| NB 93M | 4.75E-03 | 1.35E+00 | 4.71E-07 |
| NB 95 | 1.71E-16 | 6.72E-12 | 3.24E-17 |
| NB 95M | 1.73E-19 | 6.58E-14 | 3.16E-20 |
| TC 99 | 8.37E+02 | 1.44E+01 | 2.48E-05 |
| RU106 | 1.18E-01 | 3.97E+02 | 2.35E-05 |
| RH106 | 1.12E-07 | 3.97E+02 | 4.18E-03 |
| PD107 | 2.41E+02 | 1.15E-01 | 9.54E-09 |
| CD113M | 9.06E-02 | 2.06E+01 | 2.71E-05 |
| SN119M | 4.99E-07 | 2.19E-03 | 2.31E-09 |
| SN121M | 4.15E-01 | 1.62E+01 | 1.70E-05 |
| SN123 | 3.40E-10 | 2.89E-06 | 9.85E-12 |
| SN126 | 2.03E+01 | 5.91E-01 | 6.38E-07 |
| SB125 | 6.55E-01 | 6.95E+02 | 2.82E-03 |
| SB126 | 7.06E-05 | 5.85E-01 | 7.61E-06 |
| SB126M | 7.52E-09 | 5.91E-01 | 4.00E-06 |
| TE125M | 1.60E-02 | 2.88E+02 | 4.95E-04 |
| TE127 | 1.58E-13 | 4.16E-07 | 6.71E-13 |
| TE127M | 4.46E-11 | 4.21E-07 | 2.32E-13 |
| I129 | 2.27E+02 | 3.71E-02 | 2.44E-08 |
| I131 | 1.81-136 | 2.25-131 | 7.84-137 |
| XE131M | 1.70E-94 | 1.43E-89 | 2.79E-95 |
| XE133 | 2.10-208 | 3.92-203 | 5.13-209 |
| CS134 | 6.01E+00 | 7.82E+03 | 8.28E-02 |
| CS135 | 4.27E+02 | 3.77E-01 | 1.83E-07 |
| CS137 | 9.85E+02 | 8.57E+04 | 1.40E-01 |
| BA137M | 1.49E-04 | 8.01E+04 | 3.15E-01 |
| CE144 | 3.07E-02 | 9.82E+01 | 8.03E-05 |
| PR144 | 1.30E-06 | 9.82E+01 | 7.61E-04 |
| PM147 | 6.94E+00 | 6.44E+03 | 3.32E-03 |
| SM151 | 3.80E+01 | 1.03E+03 | 1.80E-03 |
| EU152 | 2.78E-02 | 5.46E+00 | 9.78E-05 |
| EU154 | 5.49E+01 | 7.08E+03 | 6.55E-02 |
| EU155 | 4.34E-02 | 5.54E+01 | 4.66E-05 |
| GD153 | 1.73E-07 | 5.09E-04 | 8.77E-10 |
| TB160 | 1.50E-17 | 1.69E-13 | 1.44E-18 |
| HO166M | 5.59E-04 | 1.00E-03 | 1.08E-08 |

Table A.5. SF(MOX) Continued

| Nuclide | Grams MTIIM(fuel) | Curies MTHM(fuel) | kW MTHM(fuel) |
|---|---|---|---|
| TL207 | 2.12E-15 | 4.03E-07 | 1.22E-12 |
| TL208 | 1.03E-13 | 3.00E-05 | 6.99E-10 |
| TL209 | 4.34E-19 | 1.78E-10 | 2.90E-15 |
| PB209 | 1.78E-15 | 8.07E-09 | 9.33E-15 |
| PB210 | 4.62E-10 | 3.75E-08 | 1.55E-15 |
| PB211 | 1.64E-14 | 4.04E-07 | 1.35E-12 |
| PB212 | 5.97E-11 | 8.33E-05 | 1.20E-10 |
| PB214 | 8.50E-15 | 2.79E-07 | 6.79E-13 |
| BI210 | 3.01E-13 | 3.73E-08 | 9.82E-14 |
| BI211 | 9.74E-16 | 4.04E-07 | 1.57E-11 |
| BI212 | 5.69E-12 | 8.33E-05 | 1.46E-09 |
| BI213 | 4.30E-16 | 8.07E-09 | 4.96E-14 |
| BI214 | 6.25E-15 | 2.79E-07 | 3.88E-12 |
| PO210 | 7.39E-12 | 3.33E-08 | 1.07E-12 |
| PO211 | 1.18E-20 | 1.21E-09 | 5.35E-14 |
| PO212 | <1.E-20 | 5.33E-05 | 2.83E-09 |
| PO213 | <1.E-20 | 7.89E-09 | 3.92E-13 |
| PO214 | 1.06E-21 | 2.79E-07 | 1.27E-11 |
| PO215 | 1.38E-20 | 4.04E-07 | 1.77E-11 |
| PO216 | <1.E-20 | 8.33E-05 | 3.41E-09 |
| PO218 | 3.85E-16 | 2.79E-07 | 1.01E-11 |
| AT217 | 4.97E-21 | 8.07E-09 | 3.38E-13 |
| RN219 | 3.14E-17 | 4.04E-07 | 1.63E-11 |
| RN220 | 9.09E-14 | 8.33E-05 | 3.16E-09 |
| RN222 | 1.81E-12 | 2.79E-07 | 9.08E-12 |
| FR221 | 4.55E-17 | 8.07E-09 | 3.00E-13 |
| FR223 | 1.51E-16 | 5.78E-09 | 1.35E-14 |
| RA223 | 7.86E-12 | 4.04E-07 | 1.40E-11 |
| RA224 | 5.20E-10 | 8.33E-05 | 2.85E-09 |
| RA225 | 2.67E-13 | 8.14E-09 | 5.35E-15 |
| RA226 | 2.82E-07 | 2.79E-07 | 7.91E-12 |
| RA228 | 2.37E-13 | 5.54E-11 | 4.27E-18 |
| AC225 | 1.39E-13 | 8.07E-09 | 2.77E-13 |
| AC227 | 5.66E-09 | 4.13E-07 | 2.08E-13 |
| AC228 | 2.47E-17 | 5.54E-11 | 2.94E-16 |
| TH227 | 1.27E-11 | 4.02E-07 | 1.38E-11 |
| TH228 | 1.01E-07 | 8.28E-05 | 2.71E-09 |
| TH229 | 3.85E-08 | 8.23E-09 | 2.49E-13 |
| TH230 | 5.16E-03 | 1.00E-04 | 2.84E-09 |
| TH231 | 2.53E-08 | 1.34E-02 | 1.06E-08 |
| TH232 | 1.35E-03 | 1.47E-10 | 3.56E-15 |
| TH234 | 1.34E-05 | 3.11E-01 | 1.11E-07 |
| PA231 | 6.00E-05 | 2.86E-06 | 8.72E-11 |
| PA233 | 1.99E-05 | 4.08E-01 | 5.51E-07 |
| PA234 | 1.57E-10 | 3.11E-04 | 2.82E-09 |
| PA234M | 4.52E-10 | 3.11E-01 | 1.60E-06 |
| U233 | 1.83E-03 | 1.73E-05 | 5.04E-10 |
| U234 | 1.70E+02 | 1.05E+00 | 3.03E-05 |
| U235 | 6.25E+03 | 1.34E-02 | 3.72E-07 |
| U236 | 4.72E+03 | 2.99E-01 | 8.12E-06 |

Table A.5. SF(MOX) Continued

| Nuclide | Grams MTIIM(fuel) | Curies MTIIM(fuel) | kW MTIIM(fuel) |
|---|---|---|---|
| U237 | 5.40E-05 | 4.41E+00 | 2.93E-06 |
| U238 | 9.32E+05 | 3.11E-01 | 7.86E-06 |
| NP237 | 5.78E+02 | 4.08E-01 | 1.20E-05 |
| NP239 | 9.19E-04 | 2.14E+02 | 2.89E-04 |
| PU238 | 6.02E+02 | 1.02E+04 | 3.37E-01 |
| PU239 | 7.64E+03 | 4.69E+02 | 1.46E-02 |
| PU240 | 4.76E+03 | 1.05E+03 | 3.27E-02 |
| PU241 | 1.81E+03 | 1.84E+05 | 7.62E-03 |
| PU242 | 2.03E+03 | 7.91E+00 | 2.34E-04 |
| AM241 | 1.24E+03 | 4.24E+03 | 1.41E-01 |
| AM242 | 1.09E-04 | 8.83E+01 | 1.18E-04 |
| AM242M | 9.08E+00 | 8.83E+01 | 2.51E-05 |
| AM243 | 1.11E+03 | 2.14E+02 | 7.80E-03 |
| CM242 | 2.19E-02 | 7.27E+01 | 2.68E-03 |
| CM244 | 4.16E+02 | 3.37E+04 | 1.18E+00 |
| CM245 | 6.61E+01 | 1.17E+01 | 3.67E-04 |
| CM246 | 5.57E+00 | 2.03E+00 | 6.66E-05 |

## APPENDIX B. PROPERTIES OF FLUIDIZED BED CALCINE

| Property Units | Fluidized Bed Calcine |
|---|---|
| Solution rate, $\frac{mg}{m^2\text{-sec}}$ | 10 to 100 |
| Corrosion to clad material, nm/sec | 0 to 10 |
| Residual nitrate and water, % | $\leq$ 0.03 |
| Volatility | 1200 K all Ru and Cs |
| Specific volume, $\frac{m^3}{mgU}$ | 0.032 to 0.040 |
| Specific area, $\frac{m^2}{kg}$ | 100 to 5000 |
| Form | Granular |
| Structural quality | Soft and crumbly |
| Porosity, % | 45 to 80 |
| Density, $\frac{kg}{m^3}$ | 2000 to 2400 |
| Coefficient of linear expansion | $8.3 \times 10^{-6}$ |
| Thermal conductivity, $\frac{W}{m\text{-}^\circ K}$ | 0.2 to 0.3 |
| Heat capacity, $\frac{J}{kg\text{-}^\circ K}$ | 650 |
| Liquidus temperature, K | 1670 |

Reference: "Determination of Performance Criteria for High-Level Solidified Nuclear Waste," U.S. Nuclear Regulatory Commission, NUREG-0279, July 1977.

# APPENDIX C. REACTOR WASTE MANAGEMENT SCHEDULE

# REACTOR WASTE MANAGEMENT SCHEDULE - Section 1

## Spent Fuel Assembly Status - No-Recycle Option

| | | | | Assemblies Discharged | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|
| | | Gigawatts | | Annual | | | Cumulative | | | MT U Discharged | |
| Year | New | Operating | Discharging | BWR | PWR | Total | BWR | PWR | Total | Annual | Cumulative |
| 1960 | 0.2 | 0.2 | 0.0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| 1961 | 0.2 | 0.4 | 0.0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| 1962 | 0.0 | 0.4 | 0.2 | 9 | 8 | 17 | 9 | 8 | 17 | 5 | 5 |
| 1963 | 0.1 | 0.5 | 0.4 | 17 | 15 | 32 | 26 | 23 | 49 | 10 | 16 |
| 1964 | 0.0 | 0.5 | 0.4 | 17 | 15 | 32 | 43 | 38 | 81 | 10 | 26 |
| 1965 | 0.0 | 0.5 | 0.5 | 22 | 19 | 41 | 65 | 57 | 122 | 13 | 39 |
| 1966 | 0.0 | 0.5 | 0.5 | 22 | 19 | 41 | 87 | 76 | 163 | 13 | 52 |
| 1967 | 0.0 | 0.5 | 0.5 | 22 | 19 | 41 | 109 | 95 | 204 | 13 | 65 |
| 1968 | 1.0 | 1.5 | 0.5 | 22 | 19 | 41 | 131 | 114 | 245 | 13 | 78 |
| 1969 | 1.3 | 2.8 | 0.5 | 22 | 19 | 41 | 153 | 133 | 286 | 13 | 91 |
| 1970 | 1.0 | 3.8 | 1.5 | 65 | 58 | 123 | 218 | 191 | 409 | 39 | 130 |
| 1971 | 3.6 | 7.4 | 2.8 | 121 | 108 | 229 | 339 | 299 | 638 | 73 | 203 |
| 1972 | 4.4 | 11.8 | 3.9 | 165 | 146 | 311 | 504 | 445 | 949 | 99 | 302 |
| 1973 | 7.0 | 18.8 | 7.4 | 321 | 285 | 606 | 825 | 730 | 1555 | 192 | 494 |
| 1974 | 10.8 | 29.6 | 11.8 | 511 | 455 | 966 | 1336 | 1185 | 2521 | 307 | 301 |
| 1975 | 6.1 | 35.7 | 18.8 | 815 | 724 | 1539 | 2151 | 1909 | 4060 | 489 | 1290 |
| 1976 | 9.3 | 44.0 | 29.6 | 1283 | 1140 | 2423 | 3434 | 3049 | 6483 | 770 | 2059 |
| 1977 | 7.0 | 51.0 | 35.7 | 1547 | 1375 | 2922 | 4981 | 4424 | 9405 | 928 | 2987 |
| 1978 | 6.0 | 57.0 | 44.0 | 1907 | 1695 | 3602 | 6888 | 6119 | 13007 | 1144 | 4131 |
| 1979 | 5.0 | 62.0 | 51.0 | 2210 | 1964 | 4174 | 9098 | 8083 | 17181 | 1326 | 5457 |
| 1980 | 9.0 | 70.0 | 57.0 | 2470 | 2196 | 4666 | 11568 | 10279 | 21847 | 1482 | 6939 |
| 1981 | 13.0 | 83.0 | 62.0 | 2687 | 2388 | 5075 | 14255 | 12667 | 26922 | 1612 | 8551 |
| 1982 | 16.0 | 99.0 | 70.0 | 3033 | 2696 | 5729 | 17288 | 15363 | 32651 | 1820 | 10371 |
| 1983 | 19.0 | 118.0 | 83.0 | 3597 | 3197 | 6794 | 20885 | 18560 | 39445 | 2158 | 12529 |
| 1984 | 17.0 | 135.0 | 99.0 | 4290 | 3813 | 8103 | 25175 | 22373 | 47548 | 2574 | 15103 |
| 1985 | 21.0 | 156.0 | 118.0 | 5113 | 4545 | 9658 | 30288 | 26918 | 57206 | 3068 | 18171 |
| 1986 | 24.0 | 180.0 | 135.0 | 5850 | 5200 | 11050 | 36138 | 32118 | 68256 | 3510 | 21681 |
| 1987 | 21.0 | 201.0 | 156.0 | 5760 | 6009 | 12769 | 42898 | 38127 | 81025 | 4056 | 25737 |
| 1988 | 21.0 | 222.0 | 180.0 | 7800 | 6933 | 14733 | 50698 | 45060 | 95758 | 4680 | 30417 |
| 1989 | 23.0 | 245.0 | 201.0 | 8710 | 7742 | 16452 | 59408 | 52802 | 112210 | 5226 | 35643 |
| 1990 | 23.2 | 268.0 | 222.2 | 9632 | 8561 | 18193 | 59040 | 61363 | 130403 | 5779 | 41422 |
| 1991 | 26.2 | 294.0 | 245.2 | 10628 | 9447 | 20075 | 79668 | 70810 | 150478 | 6377 | 47799 |
| 1992 | 26.0 | 320.0 | 268.0 | 11613 | 10323 | 21936 | 91281 | 81133 | 172414 | 6968 | 54767 |
| 1993 | 26.1 | 346.0 | 294.1 | 12746 | 11330 | 24076 | 104027 | 92463 | 196490 | 7648 | 62415 |
| 1994 | 27.0 | 373.0 | 320.0 | 13867 | 12326 | 26193 | 117894 | 104789 | 222683 | 8320 | 70735 |
| 1995 | 27.0 | 400.0 | 346.0 | 14993 | 13327 | 28320 | 132887 | 118116 | 251003 | 8996 | 79731 |
| 1996 | 24.0 | 424.0 | 373.0 | 16163 | 14367 | 30530 | 149050 | 132483 | 281533 | 9698 | 89429 |
| 1997 | 22.0 | 446.0 | 400.0 | 17333 | 15407 | 32740 | 166383 | 147890 | 314273 | 10400 | 99829 |
| 1998 | 23.0 | 468.0 | 425.0 | 18432 | 16384 | 34816 | 184815 | 164274 | 349089 | 11059 | 110888 |
| 1999 | 21.3 | 488.0 | 447.3 | 19402 | 17247 | 36649 | 204217 | 181521 | 385738 | 11642 | 122529 |
| 2000 | 20.0 | 507.0 | 469.0 | 20338 | 18079 | 38417 | 224555 | 199600 | 424155 | 12203 | 134732 |
| 2001 | 0.0 | 503.4 | 491.6 | 21357 | 18984 | 40341 | 245912 | 218584 | 464496 | 12814 | 147546 |
| 2002 | 0.0 | 499.0 | 511.4 | 22227 | 19757 | 41984 | 268139 | 238341 | 506480 | 13336 | 160882 |
| 2003 | 0.0 | 492.0 | 499.0 | 21728 | 19314 | 41042 | 289867 | 257655 | 547522 | 13037 | 173919 |
| 2004 | 0.0 | 481.2 | 492.0 | 21482 | 19095 | 40577 | 311349 | 276750 | 588099 | 12889 | 186809 |
| 2005 | 0.0 | 475.1 | 481.2 | 20944 | 18616 | 39560 | 332293 | 295366 | 627659 | 12566 | 199375 |
| 2006 | 0.0 | 466.8 | 475.1 | 20712 | 18411 | 39123 | 353005 | 313777 | 666782 | 12427 | 211802 |
| 2007 | 0.0 | 459.8 | 466.8 | 20333 | 18074 | 38407 | 373338 | 331851 | 705189 | 12200 | 224002 |
| 2008 | 0.0 | 453.8 | 459.8 | 20015 | 17791 | 37806 | 393353 | 349642 | 742995 | 12009 | 236011 |
| 2009 | 0.0 | 448.8 | 453.8 | 19740 | 17546 | 37286 | 413093 | 367188 | 780281 | 11844 | 247854 |
| 2010 | 0.0 | 440.8 | 448.8 | 19568 | 17394 | 36962 | 432661 | 384582 | 817243 | 11741 | 259595 |
| 2011 | 0.0 | 427.8 | 440.8 | 19296 | 17152 | 36448 | 451957 | 401734 | 853691 | 11578 | 271173 |
| 2012 | 0.0 | 411.8 | 427.8 | 18778 | 16692 | 35470 | 470735 | 418426 | 889161 | 11267 | 282440 |
| 2013 | 0.0 | 392.8 | 411.8 | 18130 | 16115 | 34245 | 488865 | 434541 | 923406 | 10878 | 293318 |
| 2014 | 0.0 | 375.8 | 392.8 | 17276 | 15357 | 32633 | 506141 | 449898 | 956039 | 10366 | 303683 |
| 2015 | 0.0 | 354.8 | 375.8 | 16600 | 14755 | 31355 | 522741 | 464653 | 987394 | 9960 | 313643 |
| 2016 | 0.0 | 330.8 | 354.8 | 15735 | 13986 | 29721 | 538476 | 478639 | 1017115 | 9441 | 323084 |
| 2017 | 0.0 | 309.8 | 330.8 | 14656 | 13022 | 27672 | 553126 | 491661 | 1044787 | 8790 | 331874 |
| 2018 | 0.0 | 288.8 | 309.8 | 13743 | 12213 | 25953 | 566866 | 503874 | 1070740 | 8244 | 340118 |
| 2019 | 0.0 | 265.8 | 288.8 | 12860 | 11431 | 24291 | 579726 | 515305 | 1095031 | 7716 | 347833 |
| 2020 | 0.0 | 242.6 | 265.8 | 11866 | 10548 | 22414 | 591592 | 525853 | 1117445 | 7120 | 354953 |
| 2021 | 0.0 | 216.4 | 242.6 | 10906 | 9694 | 20600 | 602498 | 535547 | 1138045 | 6543 | 361496 |
| 2022 | 0.0 | 190.4 | 216.4 | 9767 | 8682 | 18449 | 612265 | 544229 | 1156494 | 5860 | 367357 |
| 2023 | 0.0 | 164.3 | 190.4 | 8642 | 7682 | 16324 | 620907 | 551911 | 1172818 | 5185 | 372542 |
| 2024 | 0.0 | 137.3 | 164.3 | 7525 | 6689 | 14214 | 628432 | 558600 | 1187932 | 4515 | 377057 |
| 2025 | 0.0 | 110.3 | 137.3 | 6355 | 5649 | 12004 | 634787 | 564249 | 1199036 | 3813 | 380870 |
| 2026 | 0.0 | 86.3 | 110.3 | 5140 | 4569 | 9709 | 639927 | 568818 | 1208745 | 3084 | 383953 |
| 2027 | 0.0 | 64.3 | 86.3 | 4079 | 3617 | 7687 | 643997 | 572435 | 1216432 | 2442 | 386395 |
| 2028 | 0.0 | 41.3 | 64.3 | 3131 | 2783 | 5914 | 647128 | 575218 | 1222346 | 1879 | 388274 |
| 2029 | 0.0 | 20.0 | 41.3 | 2189 | 1375 | 3984 | 649237 | 577093 | 1226330 | 1266 | 389540 |
| 2030 | 0.0 | 0.0 | 20.0 | 1167 | 1037 | 2204 | 650404 | 578130 | 1228534 | 700 | 390240 |

REACTOR WASTE MANAGEMENT SCHEDULE - Section 2

Spent Fuel Assembly Status - U-Only Recycle

| Year | MT Uranium Spent Fuel | | Spent Fuel Assemblies Remaining | | |
|---|---|---|---|---|---|
| | Reprocessed | Remaining | BWR | PWR | Total |
| 1981 | 0 | 8551 | 14255 | 12667 | 26922 |
| 1982 | 300 | 10071 | 16786 | 14921 | 31707 |
| 1983 | 600 | 11629 | 19382 | 17229 | 36611 |
| 1984 | 900 | 13303 | 22172 | 19709 | 41881 |
| 1985 | 1200 | 15171 | 25286 | 22476 | 47762 |
| 1986 | 2000 | 16681 | 27802 | 24713 | 52515 |
| 1987 | 2500 | 18237 | 30396 | 27018 | 57414 |
| 1988 | 3000 | 19917 | 33196 | 29507 | 62703 |
| 1989 | 3500 | 21643 | 36072 | 32064 | 68136 |
| 1990 | 4500 | 22922 | 38204 | 33959 | 72163 |
| 1991 | 5000 | 24299 | 40499 | 35999 | 76498 |
| 1992 | 5500 | 25767 | 42946 | 38174 | 81120 |
| 1993 | 6000 | 27415 | 45691 | 40615 | 86306 |
| 1994 | 6500 | 29235 | 48725 | 43311 | 92036 |
| 1995 | 7000 | 31231 | 52051 | 46268 | 98319 |
| 1996 | 7500 | 33429 | 55715 | 49524 | 105239 |
| 1997 | 7500 | 36329 | 60548 | 53821 | 114369 |
| 1998 | 7500 | 39888 | 66480 | 59093 | 125573 |
| 1999 | 7500 | 44029 | 73382 | 65229 | 138611 |
| 2000 | 7500 | 48732 | 81221 | 72196 | 153417 |
| 2001 | 7500 | 54046 | 90077 | 80069 | 170146 |
| 2002 | 7500 | 59882 | 99804 | 88715 | 188519 |
| 2003 | 7500 | 65419 | 109032 | 96918 | 205950 |
| 2004 | 7500 | 70809 | 118014 | 104902 | 222916 |
| 2005 | 7500 | 75875 | 126458 | 112407 | 238865 |
| 2006 | 7500 | 80802 | 134670 | 119707 | 254377 |
| 2007 | 7500 | 85502 | 142503 | 126669 | 269172 |
| 2008 | 7500 | 90011 | 150018 | 133349 | 283367 |
| 2009 | 7500 | 94354 | 157257 | 139784 | 297041 |
| 2010 | 7500 | 98595 | 164325 | 146067 | 310392 |
| 2011 | 7500 | 102673 | 171122 | 152108 | 323230 |
| 2012 | 7500 | 106440 | 177400 | 157689 | 335089 |
| 2013 | 7500 | 109818 | 183029 | 162693 | 345722 |
| 2014 | 7500 | 112683 | 187806 | 166938 | 354744 |
| 2015 | 7500 | 115143 | 191905 | 170583 | 362488 |
| 2016 | 7500 | 117084 | 195140 | 173458 | 368598 |
| 2017 | 7500 | 118374 | 197290 | 175369 | 372659 |
| 2018 | 7500 | 119118 | 198529 | 176471 | 375000 |
| 2019 | 7500 | 119333 | 198889 | 176790 | 375679 |
| 2020 | 7500 | 118953 | 198255 | 176227 | 374482 |
| 2021 | 7500 | 117996 | 196661 | 174809 | 371470 |
| 2022 | 7500 | 116357 | 193928 | 172380 | 366308 |
| 2023 | 7500 | 114042 | 190070 | 168951 | 359021 |
| 2024 | 7500 | 111057 | 185095 | 164529 | 349624 |
| 2025 | 7500 | 107370 | 178949 | 159066 | 338015 |
| 2026 | 7500 | 102953 | 171589 | 152524 | 324113 |
| 2027 | 7500 | 97895 | 163159 | 145030 | 308189 |
| 2028 | 7500 | 92274 | 153790 | 136702 | 290492 |
| 2029 | 7500 | 86040 | 143399 | 127466 | 270865 |
| 2030 | 7500 | 79240 | 132066 | 117392 | 249458 |

# REACTOR WASTE MANAGEMENT SCHEDULE - Section 3

## Cumulative Wastes from Reprocessing - U-Only Recycle

| Year | Uranium (MT) | Plutonium (MT) | Plutonium (Can) | HLW ($m^3$) | HLW (Can) | Hulls ($m^3$) | Hulls (Can) | TRU - ILW ($m^3$) | TRU - ILW (Can) | TRU - LLW ($m^3$) | TRU - LLW (Can) |
|---|---|---|---|---|---|---|---|---|---|---|---|
| 1981 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| 1982 | 287 | 3 | 500 | 17 | 95 | 98 | 553 | 732 | 4136 | 292 | 1746 |
| 1983 | 859 | 9 | 1500 | 50 | 284 | 293 | 1658 | 2196 | 12407 | 875 | 5238 |
| 1984 | 1719 | 18 | 3000 | 101 | 568 | 587 | 3315 | 4392 | 24814 | 1750 | 10477 |
| 1985 | 2865 | 30 | 5000 | 168 | 947 | 978 | 5525 | 7320 | 41356 | 2916 | 17461 |
| 1986 | 4775 | 50 | 8333 | 280 | 1579 | 1630 | 9209 | 12200 | 68927 | 4860 | 29102 |
| 1987 | 7163 | 75 | 12500 | 419 | 2369 | 2445 | 13814 | 18300 | 103390 | 7290 | 43653 |
| 1988 | 10028 | 105 | 17500 | 587 | 3316 | 3423 | 19339 | 25620 | 144746 | 10206 | 61114 |
| 1989 | 13370 | 140 | 23333 | 783 | 4421 | 4564 | 25785 | 34160 | 192994 | 13608 | 81485 |
| 1990 | 17668 | 185 | 30833 | 1034 | 5843 | 6031 | 34073 | 45140 | 255028 | 17982 | 107677 |
| 1991 | 22443 | 235 | 39167 | 1314 | 7422 | 7661 | 43282 | 57340 | 323955 | 22842 | 136778 |
| 1992 | 27695 | 290 | 48333 | 1621 | 9159 | 9454 | 53412 | 70760 | 399774 | 28188 | 168790 |
| 1993 | 33425 | 350 | 58333 | 1957 | 11054 | 11410 | 64463 | 85400 | 482486 | 34020 | 203713 |
| 1994 | 39633 | 415 | 69167 | 2320 | 13106 | 13529 | 76435 | 101260 | 572090 | 40338 | 241545 |
| 1995 | 46317 | 485 | 80833 | 2711 | 15317 | 15811 | 89328 | 118340 | 668588 | 47142 | 282287 |
| 1996 | 53480 | 560 | 93333 | 3130 | 17686 | 18256 | 103141 | 136640 | 771977 | 54432 | 325940 |
| 1997 | 60642 | 635 | 105833 | 3550 | 20055 | 20701 | 116955 | 154940 | 875367 | 61722 | 369593 |
| 1998 | 67805 | 710 | 118333 | 3969 | 22423 | 23146 | 130768 | 173240 | 978757 | 69012 | 413246 |
| 1999 | 74967 | 785 | 130833 | 4388 | 24792 | 25591 | 144582 | 191540 | 1082147 | 76302 | 456898 |
| 2000 | 82130 | 860 | 143333 | 4807 | 27160 | 28036 | 158395 | 209840 | 1185537 | 83592 | 500551 |
| 2001 | 89292 | 935 | 155833 | 5227 | 29529 | 30481 | 172209 | 228140 | 1288927 | 90882 | 544204 |
| 2002 | 96455 | 1010 | 168333 | 5646 | 31898 | 32926 | 186023 | 246440 | 1392316 | 98172 | 587856 |
| 2003 | 103618 | 1085 | 180833 | 6065 | 34266 | 35371 | 199836 | 264740 | 1495706 | 105462 | 631509 |
| 2004 | 110780 | 1160 | 193333 | 6484 | 36635 | 37816 | 213650 | 283040 | 1599096 | 112752 | 675162 |
| 2005 | 117943 | 1235 | 205833 | 6904 | 39004 | 40261 | 227463 | 301340 | 1702486 | 120042 | 718814 |
| 2006 | 125105 | 1310 | 218333 | 7323 | 41372 | 42706 | 241277 | 319640 | 1805876 | 127332 | 762467 |
| 2007 | 132268 | 1385 | 230833 | 7742 | 43741 | 45151 | 255090 | 337940 | 1909266 | 134622 | 806120 |
| 2008 | 139430 | 1460 | 243333 | 8161 | 46110 | 47596 | 268904 | 356240 | 2012655 | 141912 | 849772 |
| 2009 | 146593 | 1535 | 255833 | 8581 | 48478 | 50041 | 282718 | 374540 | 2116045 | 149202 | 893425 |
| 2010 | 153755 | 1610 | 268333 | 9000 | 50847 | 52486 | 296531 | 392840 | 2219435 | 156492 | 937078 |
| 2011 | 160918 | 1685 | 280833 | 9419 | 53216 | 54931 | 310345 | 411140 | 2322825 | 163782 | 980731 |
| 2012 | 168080 | 1760 | 293333 | 9838 | 55584 | 57376 | 324158 | 429440 | 2426215 | 171072 | 1024383 |
| 2013 | 175242 | 1835 | 305833 | 10258 | 57953 | 59821 | 337972 | 447740 | 2529605 | 178362 | 1068036 |
| 2014 | 182405 | 1910 | 318333 | 10677 | 60321 | 62266 | 351785 | 466040 | 2632994 | 185652 | 1111689 |
| 2015 | 189567 | 1985 | 330833 | 11096 | 62690 | 64711 | 365599 | 484340 | 2736384 | 192942 | 1155341 |
| 2016 | 196730 | 2060 | 343333 | 11515 | 65059 | 67156 | 379412 | 502640 | 2839774 | 200232 | 1198994 |
| 2017 | 203892 | 2135 | 355833 | 11935 | 67427 | 69601 | 393226 | 520940 | 2943164 | 207522 | 1242647 |
| 2018 | 211055 | 2210 | 368333 | 12354 | 69796 | 72046 | 407040 | 539240 | 3046554 | 214812 | 1286299 |
| 2019 | 218217 | 2285 | 380833 | 12773 | 72165 | 74491 | 420853 | 557540 | 3149944 | 222102 | 1329952 |
| 2020 | 225380 | 2360 | 393333 | 13192 | 74533 | 76936 | 434667 | 575840 | 3253333 | 229392 | 1373605 |
| 2021 | 232542 | 2435 | 405833 | 13612 | 76902 | 79381 | 448480 | 594140 | 3356723 | 236682 | 1417257 |
| 2022 | 239705 | 2510 | 418333 | 14031 | 79271 | 81826 | 462294 | 612440 | 3460113 | 243972 | 1460910 |
| 2023 | 246867 | 2585 | 430833 | 14450 | 81639 | 34271 | 476107 | 630740 | 3563503 | 251262 | 1504563 |
| 2024 | 254030 | 2660 | 443333 | 14869 | 84008 | 86716 | 489921 | 649040 | 3666893 | 258552 | 1548216 |
| 2025 | 261192 | 2735 | 455833 | 15289 | 86377 | 89161 | 503734 | 667340 | 3770282 | 265842 | 1591868 |
| 2026 | 268355 | 2810 | 468333 | 15708 | 88745 | 91606 | 517548 | 685640 | 3873672 | 273132 | 1635521 |
| 2027 | 275517 | 2885 | 480833 | 16127 | 91114 | 94051 | 531362 | 703940 | 3977062 | 280422 | 1679174 |
| 2028 | 282680 | 2960 | 493333 | 16546 | 93482 | 96496 | 545175 | 722240 | 4080452 | 287712 | 1722826 |
| 2029 | 289842 | 3035 | 505833 | 16966 | 95851 | 98941 | 558989 | 740540 | 4183842 | 295002 | 1766479 |
| 2030 | 297005 | 3110 | 518333 | 17385 | 98220 | 101386 | 572802 | 758840 | 4287232 | 302292 | 1810132 |

REACTOR WASTE MANAGEMENT SCHEDULE - Section 4

Annual Wastes to Repository - U-Only Recycle

| Year | Plutonium (MT) | Plutonium (Can) | HLW ($m^3$) | HLW (Can) | Hulls ($m^3$) | Hulls (Can) | TRU - ILW ($m^3$) | TRU - ILW (Can) | TRU - LLW ($m^3$) | TRU - LLW (Can) |
|---|---|---|---|---|---|---|---|---|---|---|
| 1981 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| 1982 | 3 | 500 | 17 | 95 | 98 | 553 | 732 | 4136 | 292 | 1746 |
| 1983 | 2 | 323 | 11 | 61 | 63 | 357 | 473 | 2674 | 189 | 1129 |
| 1984 | 3 | 511 | 17 | 97 | 100 | 565 | 749 | 4229 | 298 | 1786 |
| 1985 | 5 | 815 | 27 | 154 | 159 | 900 | 1193 | 6738 | 475 | 2845 |
| 1986 | 8 | 1283 | 43 | 243 | 251 | 1417 | 1878 | 10609 | 748 | 4479 |
| 1987 | 9 | 1547 | 52 | 293 | 303 | 1710 | 2265 | 12796 | 902 | 5402 |
| 1988 | 11 | 1907 | 64 | 361 | 373 | 2107 | 2791 | 15770 | 1112 | 6658 |
| 1989 | 13 | 2210 | 74 | 419 | 432 | 2442 | 3235 | 18279 | 1289 | 7718 |
| 1990 | 15 | 2470 | 83 | 468 | 483 | 2730 | 3616 | 20430 | 1441 | 8626 |
| 1991 | 16 | 2687 | 90 | 509 | 526 | 2969 | 3933 | 22222 | 1567 | 9382 |
| 1992 | 18 | 3033 | 102 | 575 | 593 | 3352 | 4441 | 25089 | 1769 | 10593 |
| 1993 | 22 | 3597 | 121 | 682 | 704 | 3975 | 5266 | 29749 | 2098 | 12560 |
| 1994 | 26 | 4290 | 144 | 813 | 839 | 4741 | 6281 | 35483 | 2502 | 14982 |
| 1995 | 31 | 5113 | 172 | 969 | 1000 | 5651 | 7486 | 42293 | 2982 | 17857 |
| 1996 | 35 | 5850 | 196 | 1109 | 1144 | 6465 | 8564 | 48386 | 3412 | 20429 |
| 1997 | 41 | 6760 | 227 | 1281 | 1322 | 7470 | 9897 | 55913 | 3942 | 23607 |
| 1998 | 47 | 7800 | 262 | 1478 | 1526 | 8620 | 11419 | 64515 | 4549 | 27239 |
| 1999 | 52 | 8710 | 292 | 1650 | 1704 | 9625 | 12751 | 72042 | 5080 | 30417 |
| 2000 | 58 | 9632 | 323 | 1825 | 1884 | 10644 | 14101 | 79665 | 5617 | 33636 |
| 2001 | 64 | 10628 | 356 | 2014 | 2079 | 11745 | 15560 | 87909 | 6198 | 37116 |
| 2002 | 70 | 11613 | 390 | 2201 | 2272 | 12834 | 17002 | 96056 | 6773 | 40556 |
| 2003 | 76 | 12746 | 427 | 2415 | 2493 | 14085 | 18660 | 105423 | 7433 | 44511 |
| 2004 | 83 | 13867 | 465 | 2628 | 2712 | 15324 | 20301 | 114694 | 8087 | 48425 |
| 2005 | 90 | 14993 | 503 | 2841 | 2933 | 16569 | 21950 | 124013 | 8744 | 52360 |
| 2006 | 97 | 16163 | 542 | 3063 | 3162 | 17862 | 23663 | 133690 | 9426 | 56446 |
| 2007 | 104 | 17333 | 581 | 3285 | 3390 | 19155 | 25376 | 143367 | 10109 | 60532 |
| 2008 | 111 | 18432 | 618 | 3493 | 3605 | 20369 | 26984 | 152452 | 10749 | 64367 |
| 2009 | 116 | 19403 | 651 | 3677 | 3795 | 21441 | 28405 | 160482 | 11316 | 67758 |
| 2010 | 122 | 20338 | 682 | 3854 | 3979 | 22476 | 29775 | 168222 | 11861 | 71026 |
| 2011 | 128 | 21357 | 716 | 4047 | 4177 | 23601 | 31266 | 176645 | 12455 | 74582 |
| 2012 | 133 | 22227 | 745 | 4212 | 4348 | 24562 | 32540 | 183841 | 12963 | 77620 |
| 2013 | 130 | 21728 | 729 | 4117 | 4250 | 24012 | 31810 | 179719 | 12672 | 75880 |
| 2014 | 129 | 21482 | 721 | 4071 | 4202 | 23739 | 31450 | 177682 | 12528 | 75020 |
| 2015 | 117 | 19486 | 654 | 3692 | 3811 | 21533 | 28527 | 161170 | 11364 | 68048 |
| 2016 | 75 | 12500 | 419 | 2369 | 2445 | 13814 | 18300 | 103390 | 7290 | 43653 |
| 2017 | 75 | 12500 | 419 | 2369 | 2445 | 13814 | 18300 | 103390 | 7290 | 43653 |
| 2018 | 75 | 12500 | 419 | 2369 | 2445 | 13814 | 18300 | 103390 | 7290 | 43653 |
| 2019 | 75 | 12500 | 419 | 2369 | 2445 | 13814 | 18300 | 103390 | 7290 | 43653 |
| 2020 | 75 | 12500 | 419 | 2369 | 2445 | 13814 | 18300 | 103390 | 7290 | 43653 |
| 2021 | 75 | 12500 | 419 | 2369 | 2445 | 13814 | 18300 | 103390 | 7290 | 43653 |
| 2022 | 75 | 12500 | 419 | 2369 | 2445 | 13814 | 18300 | 103390 | 7290 | 43653 |
| 2023 | 75 | 12500 | 419 | 2369 | 2445 | 13814 | 18300 | 103390 | 7290 | 43653 |
| 2024 | 75 | 12500 | 419 | 2369 | 2445 | 13814 | 18300 | 103390 | 7290 | 43653 |
| 2025 | 75 | 12500 | 419 | 2369 | 2445 | 13814 | 18300 | 103390 | 7290 | 43653 |
| 2026 | 75 | 12500 | 419 | 2369 | 2445 | 13814 | 18300 | 103390 | 7290 | 43653 |
| 2027 | 75 | 12500 | 419 | 2369 | 2445 | 13814 | 18300 | 103390 | 7290 | 43653 |
| 2028 | 75 | 12500 | 419 | 2369 | 2445 | 13814 | 18300 | 103390 | 7290 | 43653 |
| 2029 | 75 | 12500 | 419 | 2369 | 2445 | 13814 | 18300 | 103390 | 7290 | 43653 |
| 2030 | 75 | 12500 | 419 | 2369 | 2445 | 13814 | 18300 | 103390 | 7290 | 43653 |

No more shipments to Repository

REACTOR WASTE MANAGEMENT SCHEDULE - Section 5

Cumulative Wastes to Repository - U-Only Recycle

| | Plutonium | | HLW | | Hulls | | TRU - ILW | | TRU - LLW | |
|---|---|---|---|---|---|---|---|---|---|---|
| Year | (MT) | (Can) | ($m^3$) | (Can) | ($m^3$) | (Can) | ($m^3$) | (Can) | ($m^3$) | (Can) |
| 1981 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| 1982 | 3 | 500 | 17 | 95 | 98 | 553 | 732 | 4136 | 292 | 1746 |
| 1983 | 5 | 823 | 28 | 156 | 161 | 910 | 1205 | 6810 | 480 | 2875 |
| 1984 | 8 | 1335 | 45 | 253 | 261 | 1475 | 1954 | 11039 | 778 | 4661 |
| 1985 | 13 | 2149 | 72 | 407 | 420 | 2375 | 3147 | 17778 | 1253 | 7506 |
| 1986 | 21 | 3432 | 115 | 650 | 671 | 3793 | 5024 | 28387 | 2002 | 11985 |
| 1987 | 30 | 4979 | 167 | 943 | 974 | 5502 | 7289 | 41182 | 2904 | 17388 |
| 1988 | 41 | 6886 | 231 | 1305 | 1347 | 7609 | 10081 | 56953 | 4016 | 24046 |
| 1989 | 55 | 9096 | 305 | 1724 | 1779 | 10051 | 13316 | 75232 | 5305 | 31764 |
| 1990 | 69 | 11566 | 388 | 2192 | 2262 | 12781 | 16932 | 95662 | 6745 | 40390 |
| 1991 | 86 | 14252 | 478 | 2701 | 2788 | 15750 | 20865 | 117884 | 8312 | 49772 |
| 1992 | 104 | 17286 | 580 | 3275 | 3381 | 19102 | 25306 | 142973 | 10081 | 60365 |
| 1993 | 125 | 20882 | 700 | 3957 | 4085 | 23077 | 30572 | 172722 | 12179 | 72926 |
| 1994 | 151 | 25172 | 844 | 4770 | 4924 | 27818 | 36852 | 208205 | 14681 | 87907 |
| 1995 | 182 | 30286 | 1016 | 5739 | 5924 | 33468 | 44338 | 250498 | 17663 | 105764 |
| 1996 | 217 | 36136 | 1212 | 6847 | 7068 | 39933 | 52903 | 298885 | 21074 | 126194 |
| 1997 | 257 | 42896 | 1439 | 8128 | 8390 | 47403 | 62799 | 354798 | 25017 | 149801 |
| 1998 | 304 | 50696 | 1700 | 9606 | 9916 | 56023 | 74218 | 419313 | 29566 | 177040 |
| 1999 | 356 | 59406 | 1992 | 11257 | 11620 | 65648 | 86970 | 491355 | 34645 | 207457 |
| 2000 | 414 | 69037 | 2316 | 13082 | 13504 | 76292 | 101071 | 571021 | 40263 | 241093 |
| 2001 | 478 | 79666 | 2672 | 15096 | 15583 | 88037 | 116631 | 658930 | 46461 | 278210 |
| 2002 | 548 | 91279 | 3061 | 17297 | 17854 | 100871 | 133632 | 754986 | 53234 | 318766 |
| 2003 | 624 | 104025 | 3489 | 19712 | 20347 | 114956 | 152292 | 860409 | 60667 | 363277 |
| 2004 | 707 | 117892 | 3954 | 22339 | 23060 | 130280 | 172593 | 975103 | 68754 | 411703 |
| 2005 | 797 | 132885 | 4457 | 25181 | 25992 | 146849 | 194543 | 1099115 | 77498 | 464062 |
| 2006 | 894 | 149048 | 4999 | 28243 | 29154 | 164711 | 218207 | 1232805 | 86925 | 520508 |
| 2007 | 998 | 166382 | 5580 | 31528 | 32544 | 183866 | 243583 | 1376172 | 97034 | 581040 |
| 2008 | 1109 | 184813 | 6199 | 35021 | 36149 | 204234 | 270566 | 1528624 | 107783 | 645407 |
| 2009 | 1225 | 204216 | 6849 | 38697 | 39945 | 225676 | 298972 | 1689106 | 119099 | 713165 |
| 2010 | 1347 | 224554 | 7532 | 42551 | 43923 | 248151 | 328747 | 1857328 | 130960 | 784191 |
| 2011 | 1475 | 245911 | 8248 | 46598 | 48100 | 271752 | 360013 | 2033973 | 143415 | 858773 |
| 2012 | 1609 | 268137 | 8993 | 50810 | 52448 | 296314 | 392553 | 2217814 | 156378 | 936393 |
| 2013 | 1739 | 289866 | 9722 | 54927 | 56698 | 320326 | 424363 | 2397533 | 169050 | 1012273 |
| 2014 | 1868 | 311348 | 10443 | 58998 | 60900 | 344066 | 455813 | 2575215 | 181578 | 1087293 |
| 2015 | 1985 | 330833 | 11096 | 62690 | 64711 | 365599 | 484340 | 2736384 | 192942 | 1155341 |
| 2016 | 2060 | 343333 | 11515 | 65059 | 67156 | 379412 | 502640 | 2839774 | 200232 | 1198994 |
| 2017 | 2135 | 355833 | 11935 | 67427 | 69601 | 393226 | 520940 | 2943164 | 207522 | 1242647 |
| 2018 | 2210 | 368333 | 12354 | 69796 | 72046 | 407040 | 539240 | 3046554 | 214812 | 1286299 |
| 2019 | 2285 | 380833 | 12773 | 72165 | 74491 | 420853 | 557540 | 3149944 | 222102 | 1329952 |
| 2020 | 2360 | 393333 | 13192 | 74533 | 76936 | 434667 | 575840 | 3253333 | 229392 | 1373605 |
| 2021 | 2435 | 405833 | 13612 | 76902 | 79381 | 448480 | 594140 | 3356723 | 236682 | 1417257 |
| 2022 | 2510 | 418333 | 14031 | 79271 | 81826 | 462294 | 612440 | 3460113 | 243972 | 1460910 |
| 2023 | 2585 | 430833 | 14450 | 81639 | 84271 | 476107 | 630740 | 3563503 | 251262 | 1504563 |
| 2024 | 2660 | 443333 | 14869 | 84008 | 86716 | 489921 | 649040 | 3666893 | 258552 | 1548216 |
| 2025 | 2735 | 455833 | 15289 | 86377 | 89161 | 503734 | 667340 | 3770282 | 265842 | 1591868 |
| 2026 | 2810 | 468333 | 15708 | 88745 | 91606 | 517548 | 685640 | 3873672 | 273132 | 1635521 |
| 2027 | 2885 | 480833 | 16127 | 91114 | 94051 | 531362 | 703940 | 3977062 | 280422 | 1679174 |
| 2028 | 2960 | 493333 | 16546 | 93482 | 96496 | 545175 | 722240 | 4080452 | 287712 | 1722826 |
| 2029 | 3035 | 505833 | 16966 | 95851 | 98941 | 558989 | 740540 | 4183842 | 295002 | 1766479 |
| 2030 | 3110 | 518333 | 17385 | 98220 | 101386 | 572802 | 758840 | 4287232 | 302292 | 1810132 |

# REACTOR WASTE MANAGEMENT SCHEDULE - Schedule 6

## MOX Fuel Status

| Year | Annual MT Reprocessed | | | Annual MT Produced | | | GW OK for MOX | Spent MOX3 Assemblies | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | | Annual | | | Cumulative | | |
| | UOX | MOX1 | MOX2 | MOX1 | MOX2 | MOX3 | | BWR | PWR | Total | BWR | PWR | Total |
| 1981 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 53.2 | 0 | 0 | 0 | 0 | 0 | 0 |
| 1982 | 300.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 54.6 | 0 | 0 | 0 | 0 | 0 | 0 |
| 1983 | 500.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 62.6 | 0 | 0 | 0 | 0 | 0 | 0 |
| 1984 | 900.0 | 0.0 | 0.0 | 255.7 | 0.0 | 0.0 | 75.6 | 0 | 0 | 0 | 0 | 0 | 0 |
| 1985 | 1200.0 | 0.0 | 0.0 | 511.3 | 0.0 | 0.0 | 91.6 | 0 | 0 | 0 | 0 | 0 | 0 |
| 1986 | 2000.0 | 0.0 | 0.0 | 767.0 | 0.0 | 0.0 | 110.6 | 0 | 0 | 0 | 0 | 0 | 0 |
| 1987 | 2500.0 | 0.0 | 0.0 | 1022.6 | 0.0 | 0.0 | 127.6 | 0 | 0 | 0 | 0 | 0 | 0 |
| 1988 | 2744.3 | 255.7 | 0.0 | 1704.3 | 0.0 | 0.0 | 148.6 | 0 | 0 | 0 | 0 | 0 | 0 |
| 1989 | 2988.7 | 511.3 | 0.0 | 2130.4 | 0.0 | 0.0 | 172.6 | 0 | 0 | 0 | 0 | 0 | 0 |
| 1990 | 3733.0 | 767.0 | 0.0 | 2338.7 | 217.9 | 0.0 | 193.6 | 0 | 0 | 0 | 0 | 0 | 0 |
| 1991 | 3977.4 | 1022.6 | 0.0 | 2546.9 | 435.7 | 0.0 | 214.6 | 0 | 0 | 0 | 0 | 0 | 0 |
| 1992 | 3795.7 | 1704.3 | 0.0 | 3181.2 | 653.6 | 0.0 | 237.6 | 0 | 0 | 0 | 0 | 0 | 0 |
| 1993 | 3869.6 | 2130.4 | 0.0 | 3389.4 | 871.4 | 0.0 | 260.8 | 0 | 0 | 0 | 0 | 0 | 0 |
| 1994 | 3943.5 | 2338.7 | 217.9 | 3234.6 | 1452.4 | 0.0 | 287.0 | 0 | 0 | 0 | 0 | 0 | 0 |
| 1995 | 4017.4 | 2546.9 | 435.7 | 3297.5 | 1815.5 | 0.0 | 313.0 | 0 | 0 | 0 | 0 | 0 | 0 |
| 1996 | 3665.2 | 3181.2 | 653.6 | 3360.5 | 1992.9 | 205.3 | 339.1 | 0 | 0 | 0 | 0 | 0 | 0 |
| 1997 | 3239.1 | 3389.4 | 871.4 | 3423.5 | 2170.4 | 410.6 | 366.1 | 0 | 0 | 0 | 0 | 0 | 0 |
| 1998 | 2813.0 | 3234.6 | 1452.4 | 3123.4 | 2710.9 | 615.9 | 393.1 | 0 | 0 | 0 | 0 | 0 | 0 |
| 1999 | 2387.0 | 3297.5 | 1815.5 | 2760.3 | 2888.4 | 821.2 | 417.1 | 0 | 0 | 0 | 0 | 0 | 0 |
| 2000 | 2146.5 | 3360.5 | 1992.9 | 2397.2 | 2756.4 | 1368.6 | 434.7 | 342 | 304 | 646 | 342 | 304 | 646 |
| 2001 | 1906.1 | 3423.5 | 2177.4 | 2034.1 | 2810.1 | 1710.8 | 450.7 | 684 | 608 | 1292 | 1026 | 912 | 1938 |
| 2002 | 1665.7 | 3123.4 | 2710.9 | 1829.2 | 2863.8 | 1978.0 | 461.2 | 1026 | 912 | 1938 | 2052 | 1824 | 3876 |
| 2003 | 1851.3 | 2760.3 | 2888.4 | 1624.3 | 2917.4 | 2945.2 | 475.1 | 1369 | 1217 | 2586 | 3421 | 3041 | 6462 |
| 2004 | 2346.4 | 2397.2 | 2756.4 | 1419.4 | 2661.7 | 2554.5 | 466.8 | 2281 | 2028 | 4309 | 5702 | 5069 | 10771 |
| 2005 | 2655.8 | 2034.1 | 2810.1 | 1577.6 | 2352.3 | 2721.7 | 459.8 | 2851 | 2534 | 5385 | 8553 | 7603 | 16156 |
| 2006 | 2807.0 | 1829.2 | 2863.8 | 1999.5 | 2042.8 | 2597.4 | 453.8 | 3130 | 2782 | 5912 | 11683 | 10385 | 22068 |
| 2007 | 2958.2 | 1624.3 | 2917.4 | 2263.2 | 1733.4 | 2648.0 | 448.8 | 3409 | 3030 | 6439 | 15092 | 13415 | 29507 |
| 2008 | 3418.9 | 1419.4 | 2661.7 | 2392.1 | 1558.8 | 2598.5 | 443.8 | 4258 | 3784 | 8042 | 19350 | 17199 | 36549 |
| 2009 | 3570.1 | 1577.6 | 2352.3 | 2520.9 | 1384.2 | 2749.1 | 427.8 | 4536 | 4032 | 8568 | 23886 | 21231 | 45117 |
| 2010 | 3457.6 | 1999.5 | 2042.8 | 2913.5 | 1209.6 | 2508.1 | 411.8 | 4329 | 3848 | 8177 | 28215 | 25079 | 53294 |
| 2011 | 3503.4 | 2263.2 | 1733.4 | 3042.3 | 1344.4 | 2216.6 | 392.8 | 4413 | 3923 | 8336 | 32628 | 29002 | 61630 |
| 2012 | 3549.1 | 2392.1 | 1558.8 | 2946.5 | 1704.0 | 1925.0 | 375.8 | 4498 | 3998 | 8496 | 37126 | 33000 | 70126 |
| 2013 | 3594.9 | 2520.9 | 1384.2 | 2985.5 | 1929.7 | 1633.4 | 354.8 | 4582 | 4073 | 8655 | 41708 | 37073 | 78781 |
| 2014 | 3376.9 | 2913.5 | 1209.6 | 3024.5 | 2038.5 | 1468.9 | 330.8 | 4180 | 3716 | 7896 | 45888 | 40789 | 86677 |
| 2015 | 3113.2 | 3042.3 | 1344.4 | 3063.4 | 2148.3 | 1304.3 | 309.8 | 3694 | 3284 | 6978 | 49582 | 44073 | 93655 |
| 2016 | 2571.1 | 2946.5 | 1704.0 | 2877.7 | 2482.8 | 1139.8 | 288.8 | 3208 | 2952 | 6060 | 52790 | 46925 | 99715 |
| 2017 | 1484.3 | 2985.5 | 1929.7 | 2653.0 | 2592.6 | 1266.9 | 265.8 | 2722 | 2420 | 5142 | 55512 | 49345 | 104857 |
| 2018 | 530.6 | 3024.5 | 2038.5 | 2191.0 | 2518.9 | 1605.6 | 242.6 | 2448 | 2176 | 4624 | 57960 | 51521 | 109481 |
| 2019 | 0.0 | 2637.3 | 2148.3 | 1264.9 | 2544.2 | 1817.4 | 216.4 | 2174 | 1932 | 4106 | 60134 | 53453 | 113587 |
| 2020 | 0.0 | 1443.7 | 2482.8 | 452.2 | 2577.4 | 1920.9 | 190.4 | 1900 | 1689 | 3589 | 62034 | 55142 | 117176 |
| 2021 | 0.0 | 498.4 | 2592.6 | 0.0 | 2247.5 | 2024.3 | 164.3 | 2111 | 1877 | 3988 | 64145 | 57019 | 121164 |
| 2022 | 0.0 | 0.0 | 2381.2 | 0.0 | 1230.2 | 2339.6 | 137.3 | 2676 | 2379 | 5055 | 66821 | 59398 | 126219 |
| 2023 | 0.0 | 0.0 | 1774.2 | 0.0 | 424.8 | 2443.0 | 110.3 | 3029 | 2692 | 5721 | 69850 | 62090 | 131940 |
| 2024 | 0.0 | 0.0 | 1139.5 | 0.0 | 0.0 | 2243.8 | 86.3 | 3201 | 2846 | 6047 | 73051 | 64936 | 137987 |
| 2025 | 0.0 | 0.0 | 551.8 | 0.0 | 0.0 | 1671.8 | 64.3 | 3374 | 2999 | 6373 | 76425 | 67935 | 144360 |
| 2026 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 1073.8 | 41.3 | 3899 | 3466 | 7365 | 80324 | 71401 | 151725 |
| 2027 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 520.0 | 20.0 | 4072 | 3619 | 7691 | 84396 | 75020 | 159416 |
| 2028 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 3740 | 3324 | 7064 | 98136 | 78344 | 166480 |
| 2029 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 2786 | 2477 | 5263 | 90922 | 80821 | 171743 |
| 2030 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 1790 | 1591 | 3381 | 92712 | 82412 | 175124 |
| 2031 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 867 | 770 | 1637 | 93579 | 83182 | 176761 |
| Totals | 99181.1 | 76596.7 | 59585.5 | 84519.5 | 65273.7 | 56147.9 | | | | | | | |

# REACTOR WASTE MANAGEMENT SCHEDULE - Section 7

## Cumulative Wastes from Reprocessing - MOX Recycle Option

| Year | HLW - UOX | | HLW - MOX | | HULLS | | TRU - ILW | | TRU - LLW | |
|---|---|---|---|---|---|---|---|---|---|---|
| | ($m^3$) | (Can) | ($m^3$) | (Can) | ($m^3$) | (Can) | ($m^3$) | (Can) | ($m^3$) | (Can) |
| 1981 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| 1982 | 17 | 96 | 0 | 0 | 98 | 553 | 740 | 4183 | 2490 | 14913 |
| 1983 | 51 | 287 | 0 | 0 | 293 | 1658 | 2221 | 12548 | 7471 | 44738 |
| 1984 | 102 | 575 | 0 | 0 | 587 | 3315 | 4442 | 25096 | 14943 | 89477 |
| 1985 | 170 | 958 | 0 | 0 | 978 | 5525 | 7403 | 41825 | 24904 | 149128 |
| 1986 | 283 | 1596 | 0 | 0 | 1630 | 9209 | 12339 | 69711 | 41507 | 248546 |
| 1987 | 424 | 2394 | 0 | 0 | 2445 | 13814 | 18508 | 104566 | 62261 | 372819 |
| 1988 | 579 | 3270 | 14 | 82 | 3423 | 19339 | 26252 | 148314 | 88309 | 528796 |
| 1989 | 748 | 4224 | 43 | 245 | 4564 | 25795 | 35569 | 200953 | 119652 | 716478 |
| 1990 | 959 | 5416 | 97 | 490 | 6031 | 34073 | 47694 | 269457 | 160440 | 960718 |
| 1991 | 1183 | 6685 | 144 | 816 | 7661 | 43282 | 61393 | 346852 | 206523 | 1236663 |
| 1992 | 1398 | 7897 | 241 | 1360 | 9454 | 53412 | 77232 | 436341 | 259807 | 1555728 |
| 1993 | 1616 | 9132 | 361 | 2040 | 11410 | 64463 | 94873 | 536004 | 319148 | 1911064 |
| 1994 | 1839 | 10391 | 506 | 2856 | 13529 | 76435 | 114484 | 646804 | 385120 | 2306110 |
| 1995 | 2066 | 11673 | 674 | 3808 | 15811 | 89328 | 136067 | 768742 | 457725 | 2740868 |
| 1996 | 2273 | 12843 | 891 | 5032 | 18256 | 103141 | 160189 | 905020 | 538868 | 3226752 |
| 1997 | 2456 | 13877 | 1131 | 6393 | 20701 | 116955 | 185047 | 1045465 | 622491 | 3727492 |
| 1998 | 2615 | 14775 | 1396 | 7889 | 23146 | 130768 | 210928 | 1191685 | 709553 | 4248823 |
| 1999 | 2750 | 15537 | 1685 | 9521 | 25591 | 144582 | 237660 | 1342714 | 799480 | 4787303 |
| 2000 | 2871 | 16222 | 1988 | 11230 | 28036 | 158395 | 264852 | 1496336 | 890949 | 5335026 |
| 2001 | 2979 | 16831 | 2304 | 13015 | 30481 | 172209 | 292502 | 1652551 | 983963 | 5891994 |
| 2002 | 3073 | 17362 | 2633 | 14878 | 32926 | 186023 | 320895 | 1812967 | 1079478 | 6463939 |
| 2003 | 3178 | 17953 | 2952 | 16681 | 35371 | 199836 | 349181 | 1972774 | 1174630 | 7033712 |
| 2004 | 3310 | 19702 | 3244 | 18326 | 37816 | 213650 | 376705 | 2128275 | 1267219 | 7588136 |
| 2005 | 3460 | 19550 | 3517 | 19872 | 40261 | 227463 | 403859 | 2281589 | 1358565 | 8135117 |
| 2006 | 3619 | 20446 | 3783 | 21370 | 42706 | 241277 | 430854 | 2434205 | 1449375 | 8678895 |
| 2007 | 3786 | 21391 | 4039 | 22820 | 45151 | 255090 | 457690 | 2585822 | 1539651 | 9219469 |
| 2008 | 3979 | 22482 | 4270 | 24123 | 47596 | 268904 | 483713 | 2732844 | 1627192 | 9743662 |
| 2009 | 4181 | 23621 | 4492 | 25377 | 50041 | 282718 | 509293 | 2877360 | 1713239 | 10258918 |
| 2010 | 4376 | 24725 | 4720 | 26667 | 52486 | 296531 | 534779 | 3021351 | 1798974 | 10772301 |
| 2011 | 4574 | 25843 | 4946 | 27943 | 54931 | 310345 | 559962 | 3163628 | 1883689 | 11279573 |
| 2012 | 4775 | 26976 | 5169 | 29204 | 57376 | 324158 | 584947 | 3304787 | 1967738 | 11782862 |
| 2013 | 4978 | 28124 | 5390 | 30451 | 59821 | 337972 | 609735 | 3444830 | 2051122 | 12282169 |
| 2014 | 5169 | 29202 | 5623 | 31767 | 62266 | 351785 | 634675 | 3585737 | 2135021 | 12784557 |
| 2015 | 5345 | 30196 | 5871 | 33167 | 64711 | 365599 | 660072 | 3729222 | 2220455 | 13296140 |
| 2016 | 5490 | 31016 | 6133 | 34652 | 67065 | 378900 | 685415 | 3872399 | 2305706 | 13806623 |
| 2017 | 5574 | 31490 | 6411 | 36220 | 69151 | 390584 | 709253 | 4007077 | 2385896 | 14286801 |
| 2018 | 5604 | 31659 | 6697 | 37837 | 70975 | 400986 | 731388 | 4132137 | 2460359 | 14732691 |
| 2019 | 5604 | 31659 | 6967 | 39364 | 72535 | 409801 | 751247 | 4244336 | 2527155 | 15132722 |
| 2020 | 5604 | 31659 | 7189 | 40618 | 73815 | 417032 | 768106 | 4339581 | 2583875 | 15472308 |
| 2021 | 5604 | 31659 | 7364 | 41604 | 74822 | 422726 | 781878 | 4417388 | 2630203 | 15749719 |
| 2022 | 5604 | 31659 | 7498 | 42364 | 75599 | 427111 | 792788 | 4479026 | 2666904 | 15969484 |
| 2023 | 5604 | 31659 | 7599 | 42931 | 76177 | 430379 | 800916 | 4524951 | 2694249 | 16133226 |
| 2024 | 5604 | 31659 | 7663 | 43294 | 76549 | 432478 | 806137 | 4554449 | 2711812 | 16238397 |
| 2025 | 5604 | 31659 | 7694 | 43471 | 76728 | 433494 | 808666 | 4568734 | 2720318 | 16289328 |

# REACTOR WASTE MANAGEMENT SCHEDULE - Section 8

## Annual Repository Schedule: MOX Recycle Option

| Year | HLW - UOX ($m^3$) | HLW - UOX (Can) | HLW - MOX ($m^3$) | HLW - MOX (Can) | HULLS ($m^3$) | HULLS (Can) | TRU - ILW ($m^3$) | TRU - ILW (Can) | TRU - LLW ($m^3$) | TRU - LLW (Can) |
|---|---|---|---|---|---|---|---|---|---|---|
| 1981 | 0 | 0 | 0 | 0 | 3 | 0 | 0 | 0 | 0 | 0 |
| 1982 | 17 | 96 | 0 | 0 | 98 | 553 | 740 | 4183 | 2490 | 14913 |
| 1983 | 11 | 62 | 0 | 0 | 63 | 357 | 479 | 2705 | 1610 | 9644 |
| 1984 | 17 | 98 | 0 | 0 | 100 | 565 | 757 | 4277 | 2547 | 15251 |
| 1985 | 28 | 156 | 0 | 0 | 159 | 900 | 1206 | 6815 | 4058 | 24298 |
| 1986 | 43 | 246 | 0 | 0 | 251 | 1417 | 1899 | 10730 | 6389 | 38256 |
| 1987 | 52 | 296 | 0 | 0 | 303 | 1713 | 2291 | 12941 | 7705 | 46140 |
| 1988 | 65 | 365 | 0 | 0 | 373 | 2107 | 2823 | 15950 | 9497 | 56867 |
| 1989 | 75 | 423 | 0 | 0 | 432 | 2442 | 3272 | 18487 | 11008 | 65914 |
| 1990 | 84 | 473 | 0 | 0 | 483 | 2730 | 3657 | 20662 | 12303 | 73669 |
| 1991 | 91 | 515 | 0 | 0 | 526 | 2969 | 3978 | 22475 | 13382 | 80131 |
| 1992 | 103 | 581 | 0 | 0 | 593 | 3352 | 4491 | 25375 | 15109 | 90471 |
| 1993 | 122 | 689 | 0 | 0 | 704 | 3975 | 5325 | 30087 | 17915 | 107273 |
| 1994 | 145 | 822 | 0 | 0 | 839 | 4741 | 6352 | 35887 | 21368 | 127952 |
| 1995 | 173 | 979 | 0 | 0 | 1000 | 5651 | 7571 | 42775 | 25469 | 152508 |
| 1996 | 198 | 1120 | 0 | 0 | 1144 | 6465 | 8662 | 48937 | 29138 | 174479 |
| 1997 | 229 | 1295 | 0 | 0 | 1322 | 7470 | 10009 | 56549 | 33671 | 201621 |
| 1998 | 264 | 1494 | 14 | 82 | 1609 | 9091 | 12520 | 70735 | 42117 | 252137 |
| 1999 | 295 | 1668 | 29 | 163 | 1870 | 10567 | 14838 | 83833 | 49916 | 298896 |
| 2000 | 327 | 1845 | 43 | 245 | 2134 | 12056 | 17174 | 97028 | 57773 | 345943 |
| 2001 | 360 | 2036 | 58 | 326 | 2412 | 13629 | 19621 | 110851 | 66003 | 395227 |
| 2002 | 372 | 2104 | 96 | 544 | 2705 | 15282 | 22742 | 128488 | 76504 | 458108 |
| 2003 | 105 | 591 | 120 | 680 | 1298 | 7334 | 12660 | 71524 | 42587 | 255010 |
| 2004 | 133 | 749 | 144 | 816 | 1598 | 9030 | 15670 | 88534 | 52715 | 315657 |
| 2005 | 150 | 848 | 169 | 952 | 1838 | 10385 | 18223 | 102955 | 61301 | 367075 |
| 2006 | 159 | 896 | 217 | 1224 | 2165 | 12233 | 22003 | 124313 | 74018 | 443225 |
| 2007 | 167 | 944 | 241 | 1360 | 2353 | 13296 | 24166 | 136529 | 81292 | 486778 |
| 2008 | 193 | 1091 | 265 | 1496 | 2643 | 14929 | 27376 | 154666 | 92092 | 551446 |
| 2009 | 202 | 1140 | 289 | 1632 | 2831 | 15993 | 29652 | 167525 | 99748 | 597293 |
| 2010 | 195 | 1104 | 302 | 1709 | 2872 | 15228 | 30427 | 171902 | 102354 | 612898 |
| 2011 | 198 | 1118 | 316 | 1786 | 2966 | 16755 | 31592 | 178484 | 106273 | 636357 |
| 2012 | 201 | 1133 | 330 | 1862 | 3059 | 17283 | 33041 | 186675 | 111150 | 665570 |
| 2013 | 203 | 1148 | 319 | 1803 | 3013 | 17025 | 32588 | 184115 | 109626 | 656443 |
| 2014 | 191 | 1078 | 291 | 1645 | 2781 | 15712 | 30067 | 169869 | 101144 | 605650 |
| 2015 | 176 | 994 | 274 | 1546 | 2594 | 14656 | 28283 | 159791 | 95143 | 569719 |
| 2016 | 145 | 821 | 265 | 1498 | 2368 | 13379 | 26413 | 149226 | 88852 | 532049 |
| 2017 | 84 | 474 | 257 | 1450 | 1964 | 11099 | 23199 | 131067 | 78040 | 467304 |
| 2018 | 30 | 169 | 231 | 1303 | 1503 | 8494 | 18895 | 106754 | 63563 | 380619 |
| 2019 | 0 | 0 | 222 | 1254 | 1281 | 7238 | 16769 | 94741 | 56411 | 337789 |
| 2020 | 0 | 0 | 228 | 1290 | 1318 | 7445 | 16954 | 95784 | 57032 | 341507 |
| 2021 | 0 | 0 | 226 | 1276 | 1303 | 7361 | 16537 | 93432 | 55631 | 333122 |
| 2022 | 0 | 0 | 223 | 1261 | 1288 | 7277 | 16227 | 91677 | 54587 | 326865 |
| 2023 | 0 | 0 | 221 | 1247 | 1273 | 7193 | 15916 | 89923 | 53542 | 320609 |
| 2024 | 0 | 0 | 233 | 1316 | 1344 | 7594 | 16607 | 93826 | 55866 | 334525 |
| 2025 | 0 | 0 | 248 | 1403 | 1430 | 8080 | 17714 | 100081 | 59590 | 356826 |
| 2026 | 0 | 0 | 263 | 1484 | 1516 | 8565 | 18998 | 107331 | 63907 | 382675 |
| 2027 | 0 | 0 | 278 | 1569 | 1602 | 9051 | 20175 | 113984 | 67868 | 406396 |
| 2028 | 0 | 0 | 286 | 1616 | 1651 | 9325 | 20826 | 117663 | 70059 | 419513 |
| 2029 | 0 | 0 | 270 | 1528 | 1560 | 8814 | 19859 | 112198 | 66805 | 400031 |
| 2030 | 0 | 0 | 222 | 1253 | 1280 | 7232 | 16858 | 95245 | 56711 | 339586 |
| 2031 | 0 | 0 | 175 | 987 | 1008 | 5693 | 13772 | 77807 | 46328 | 277411 |
| 2032 | 0 | 0 | 135 | 760 | 776 | 4386 | 10910 | 61638 | 36701 | 219765 |
| 2033 | 0 | 0 | 100 | 566 | 578 | 3268 | 8129 | 45925 | 27345 | 163742 |
| 2034 | 0 | 0 | 64 | 364 | 371 | 2099 | 5221 | 29498 | 17564 | 105171 |
| 2035 | 0 | 0 | 31 | 176 | 180 | 1016 | 2528 | 14285 | 8505 | 50930 |

# REACTOR WASTE MANAGEMENT SCHEDULE - Section 9

## Cumulative Repository Schedule: MOX Recycle Option

| Year | HLW - UOX ($m^3$) | HLW - UOX (Can) | HLW - MOX ($m^3$) | HLW - MOX (Can) | HULLS ($m^3$) | HULLS (Can) | TRU - ILW ($m^3$) | TRU - ILW (Can) | TRU - LLW ($m^3$) | TRU - LLW (Can) |
|---|---|---|---|---|---|---|---|---|---|---|
| 1981 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| 1982 | 17 | 96 | 0 | 0 | 98 | 553 | 740 | 4183 | 2490 | 14913 |
| 1983 | 28 | 158 | 0 | 0 | 161 | 913 | 1219 | 6887 | 4101 | 24556 |
| 1984 | 45 | 256 | 0 | 0 | 261 | 1475 | 1976 | 11165 | 6648 | 39807 |
| 1985 | 73 | 412 | 0 | 0 | 420 | 2375 | 3182 | 17980 | 10706 | 64105 |
| 1986 | 116 | 657 | 0 | 0 | 671 | 3793 | 5082 | 28710 | 17094 | 102361 |
| 1987 | 169 | 954 | 0 | 0 | 974 | 5502 | 7372 | 41651 | 24800 | 148501 |
| 1988 | 233 | 1319 | 0 | 0 | 1347 | 7609 | 10195 | 57601 | 34297 | 205369 |
| 1989 | 308 | 1742 | 0 | 0 | 1779 | 10051 | 13468 | 76088 | 45304 | 271283 |
| 1990 | 392 | 2215 | 0 | 0 | 2262 | 12781 | 17125 | 96750 | 57607 | 344952 |
| 1991 | 483 | 2730 | 0 | 0 | 2788 | 15750 | 21103 | 119225 | 70989 | 425083 |
| 1992 | 586 | 3311 | 0 | 0 | 3381 | 19102 | 25594 | 144600 | 86098 | 515554 |
| 1993 | 708 | 3999 | 0 | 0 | 4085 | 23077 | 30920 | 174687 | 104012 | 622827 |
| 1994 | 853 | 4821 | 0 | 0 | 4924 | 27818 | 37272 | 210574 | 125380 | 750778 |
| 1995 | 1027 | 5800 | 0 | 0 | 5924 | 33468 | 44843 | 253348 | 150849 | 903286 |
| 1996 | 1225 | 6921 | 0 | 0 | 7068 | 39933 | 53504 | 302285 | 179987 | 1077766 |
| 1997 | 1454 | 9216 | 0 | 0 | 3390 | 47403 | 63514 | 358835 | 213657 | 1279386 |
| 1998 | 1719 | 9710 | 14 | 82 | 9999 | 56494 | 76034 | 429569 | 255774 | 1531583 |
| 1999 | 2014 | 11378 | 43 | 245 | 11870 | 67061 | 90972 | 513402 | 305690 | 1830479 |
| 2000 | 2340 | 13222 | 87 | 490 | 14004 | 79117 | 108046 | 610430 | 363463 | 2176423 |
| 2001 | 2701 | 15258 | 144 | 816 | 16416 | 92746 | 127667 | 721281 | 429466 | 2571650 |
| 2002 | 3073 | 17362 | 241 | 1360 | 19121 | 108028 | 150409 | 849769 | 505970 | 3029759 |
| 2003 | 3178 | 17953 | 361 | 2040 | 20419 | 115361 | 163069 | 921292 | 548556 | 3284769 |
| 2004 | 3310 | 18702 | 506 | 2856 | 22017 | 124392 | 178739 | 1009826 | 601271 | 3600425 |
| 2005 | 3460 | 19550 | 674 | 3808 | 23855 | 134776 | 196962 | 1112781 | 662572 | 3967500 |
| 2006 | 3619 | 20446 | 891 | 5032 | 26021 | 147009 | 218966 | 1237094 | 736591 | 4410724 |
| 2007 | 3786 | 21391 | 1131 | 6393 | 28374 | 160306 | 243131 | 1373622 | 817083 | 4897502 |
| 2008 | 3979 | 22482 | 1396 | 7889 | 31017 | 175235 | 270507 | 1528289 | 909974 | 5448949 |
| 2009 | 4181 | 23621 | 1685 | 9521 | 33847 | 191228 | 300159 | 1695814 | 1009722 | 6046242 |
| 2010 | 4376 | 24725 | 1988 | 11230 | 36720 | 207456 | 330586 | 1867716 | 1112076 | 6659140 |
| 2011 | 4574 | 25843 | 2304 | 13015 | 39685 | 224211 | 362177 | 2046200 | 1218350 | 7295507 |
| 2012 | 4775 | 26976 | 2633 | 14879 | 42744 | 241494 | 395219 | 2232875 | 1329500 | 7961077 |
| 2013 | 4978 | 28124 | 2952 | 16681 | 45758 | 258519 | 427807 | 2416991 | 1439126 | 8617520 |
| 2014 | 5169 | 29202 | 3244 | 18326 | 48539 | 274230 | 457874 | 2586860 | 1540269 | 9223170 |
| 2015 | 5345 | 30196 | 3517 | 19872 | 51133 | 288886 | 486157 | 2746651 | 1635412 | 9792889 |
| 2016 | 5490 | 31016 | 3783 | 21370 | 53501 | 302265 | 512570 | 2895877 | 1724265 | 10324938 |
| 2017 | 5574 | 31490 | 4039 | 22820 | 55465 | 313364 | 535769 | 3026944 | 1802304 | 10792242 |
| 2018 | 5604 | 31659 | 4270 | 24123 | 56969 | 321858 | 554664 | 3133698 | 1865868 | 11172861 |
| 2019 | 5604 | 31659 | 4492 | 25377 | 58250 | 329096 | 571434 | 3228439 | 1922279 | 11510650 |
| 2020 | 5604 | 31659 | 4720 | 26667 | 59568 | 336542 | 588387 | 3324223 | 1979310 | 11852157 |
| 2021 | 5604 | 31659 | 4946 | 27943 | 60871 | 343903 | 604925 | 3417655 | 2034942 | 12185279 |
| 2022 | 5604 | 31659 | 5169 | 29204 | 62159 | 351179 | 621152 | 3509332 | 2089528 | 12512145 |
| 2023 | 5604 | 31659 | 5390 | 30451 | 63432 | 358372 | 637068 | 3599255 | 2143070 | 12832754 |
| 2024 | 5604 | 31659 | 5623 | 31767 | 64776 | 365966 | 653675 | 3693080 | 2198936 | 13167279 |
| 2025 | 5604 | 31659 | 5871 | 33167 | 66206 | 374045 | 671389 | 3793161 | 2258526 | 13524105 |
| 2026 | 5604 | 31659 | 6133 | 34652 | 67722 | 382611 | 690387 | 3900491 | 2322432 | 13906781 |
| 2027 | 5604 | 31659 | 6411 | 36220 | 69324 | 391662 | 710562 | 4014475 | 2390301 | 14313177 |
| 2028 | 5604 | 31659 | 6697 | 37837 | 70975 | 400986 | 731388 | 4132137 | 2460359 | 14732691 |
| 2029 | 5604 | 31659 | 6967 | 39364 | 72535 | 409801 | 751247 | 4244336 | 2527165 | 15132722 |
| 2030 | 5604 | 31659 | 7189 | 40618 | 73815 | 417032 | 768106 | 4339581 | 2583875 | 15472308 |
| 2031 | 5604 | 31659 | 7364 | 41604 | 74822 | 422726 | 781878 | 4417388 | 2630203 | 15749719 |
| 2032 | 5604 | 31659 | 7498 | 42364 | 75599 | 427111 | 792788 | 4479026 | 2666904 | 15969484 |
| 2033 | 5604 | 31659 | 7599 | 42931 | 76177 | 430379 | 800916 | 4524951 | 2694249 | 16133226 |
| 2034 | 5604 | 31659 | 7663 | 43294 | 76549 | 432478 | 806137 | 4554449 | 2711812 | 16238397 |
| 2035 | 5604 | 31659 | 7694 | 43471 | 76728 | 433494 | 808666 | 4568734 | 2720318 | 16289328 |

# APPENDIX D. GEOLOGICAL CONSIDERATIONS FOR DEEP STORAGE

Except for surface storage of spent fuel, all options considered in this report would require either retrievable or nonretrievable deep geological storage of radioactive wastes. Deep geological formations are being considered for repositories to ensure that radioactive wastes are contained, isolated, and secured for as long as they might pose a radiation threat to the biosphere. Nonretrievable disposal is based upon the supposition that nuclear waste can be disposed of in a formation so stable and isolated that active surveillance and management would not be needed after waste emplacement. The rock of a geological repository would protect the waste from exposure and leaching, dissipate the decay heat, and contain the radioactivity emitted by the wastes. Formations used for retrievable storage would have to have all the qualities necessary for permanent disposal plus a structural competence that would permit access to and removal of the wastes up to 25 years after emplacement.

## D.1 FORMATION DESCRIPTIONS

When geological containment is to be permanent, human-engineered barriers cannot be depended upon to maintain their integrity; instead, the repository site must be selected so that continued isolation of the wastes would be provided by the surrounding geologic materials.* The selection of a host formation and repository design must be guided by the properties of different rock types, the relative hazards of different locations, and the characteristics of the wastes. Long-term isolation and containment of spent fuel, high-level waste, intermediate-level transuranic waste, hulls, and plutonium may be accomplished by burying canisters in deep holes within the host formation and backfilling the storage rooms with mined rock. If the host rock were plastic like salt, it would creep and recrystalize to isolate the wastes. If the host rock were crystalline, careful analysis should be made to ensure that any rock fracturing around and over the mined openings would not breach the long-term integrity of the formation. A rock of uniform composition is more likely to provide long-term integrity. The repository should be deep enough to be isolated from such surface phenomena as weathering, erosion, biological processes, and circulating water. Estimates of erosion rate indicate that about 250 m (820 ft) would be a reasonably safe minimum depth for the upper horizon of a repository formation.[1] The maximum depth would be limited by economics, the geothermal gradient, and local conditions of the formation. Plastic rocks such as salt and shale may flow under extreme heat and pressure and are impractical for use as a repository at depths greater than 1500 m (4900 ft). Openings in brittle rocks can be maintained at greater depths.

The required thickness and lateral extent of a potential host formation would have to be determined for each specific repository design and for each formation. The formation would have to be large enough that the rock's qualities would protect the repository, even under adverse conditions. For example, if the host rock were plastic, like salt, there should be sufficient rock around the repository to allow fractures to seal.

Stored radionuclides are susceptible to being leached by groundwater, and permanent isolation of wastes from circulating water is difficult to guarantee. If aquifers were located stratigraphically near the repository, the host rock would have to be especially thick. It has been suggested that stored waste should be separated from the nearest aquifer by at least 100 m (330 ft) of rock.[1] Formation thicknesses proposed for model repositories in salt range from 60 m (200 ft)[2] to at least 350 m (1150 ft).[3] Highly permeable formations should be avoided because they may become aquifers if the climate changes or a recharge source is provided. Some rocks likely to have low permeabilities, and which therefore might make good repository hosts, include salt, carbonates, shales, and massive igneous or metamorphic rocks.

The host formations also would have to be sufficiently impermeable to prevent, or at least greatly restrict, the release of gases from the repository. It is assumed that all wastes disposed in deep geological repositories would be in solid form, but some radioactive gases would

*Herein referred to interchangeably as "host" or "repository" rock or formation.

be produced from decay of spent fuel and radiolysis. These gases include krypton-85 (Kr-85), carbon-14 (C-14), tritium (H-3), xenons, and iodine-129 (I-129). Waste canisters cannot be expected to be permanent barriers to the release of such gases. Only the impermeability of the host rock and overlying strata would restrict the release of these gases from a sealed repository to the atmosphere. Diffusion through 250 m to 1300 m (820 ft to 4300 ft) of overburden would greatly dilute any gases that did escape.

Shielding properties of the host rock also would be important, particularly when miners and waste transport operators were in the mine. During handling, canisters containing spent fuel, high-level wastes, hulls, plutonium, and intermediate-level transuranic wastes would be shielded by casks, transfer galleries, and shielded transporters. After emplacement, the shielding would consist of the host rock and a concrete plug. Several meters of most earth materials would provide sufficient shielding.

Heat released by the stored nuclear wastes might affect physical and chemical properties of the wastes and of the surrounding rock. Thermal effects that would influence the allowable temperature rises and heat release rates include (1) the thermal stability of wastes, (2) the thermal stability of the host formation, (3) migration of water contained in the pores or small cavities of the formation, (4) structural integrity of the entire formation, (5) temperature rise in any nearby freshwater aquifer, (6) heating of the earth's surface, and (7) temperature increases beyond the boundary of the disposal area.[1]

The most important variable affecting the temperature distribution differences among various potential host rocks would be the thermal conductivity of the geologic medium. The thermal conductivity of salt may be a factor of two higher than that of some common rocks composed of other minerals;[4] hence, approximately twice the temperature rise could be expected in other rocks for the same thermal output from the waste. Water, brine, and gases trapped in the pores of repository rocks might also have a strong influence on thermal conductivity. When the water contained in clays, shales, or mudstones is released by heating, the thermal conductivity may be reduced by factors of two or three.[1] Further evaluation of the thermal properties of these detrital rocks would be needed before they could be used as host formations for deep geological storage of radioactive wastes.

## D.2 STABILITY

A geological formation in which nuclear wastes are placed must be able to absorb the thermal, radiological, and chemical perturbations caused by the wastes, both near-field and far-field. Repository stability is directly affected by these interrelationships.

At least six kinds of incidents could have the potential for breaching a sealed repository. The results could include leaching of the wastes or even surface exposure. The types of incidents to be described include reactions of the host formation to the wastes, faulting, volcanism, erosion, cratering, and drilling. The following discussions will describe ways to avoid some of the less desirable situations.

### D.2.1 Thermal Effects

The greatest near-field impact of temperature increases probably would be on the pore water in the host rock. Pores in rock salt generally contain brine. Experimental work has shown that the creation of a thermal gradient across a salt mass causes these fluid inclusions to migrate. Those containing less than 10% vapor move upgradient. The higher temperature on one side of these fluid inclusions increases dissolution, while salt precipitation occurs on the cooler side.[5] Brine inclusions consisting of more than 10% vapor tend to migrate away from a heat source as evaporation on the hotter side of the inclusion is paired with condensation and dissolution on the cooler side.[6] Migration of both types of cavities is apparently proportional to inclusion size[5] and requires a heat gradient.[1] The environmental effects of the brine inclusions would be numerous and varied. Brine inclusions that migrated all the way to a waste canister would produce an undesired water source in the storage area. The presence of the brine would accelerate canister corrosion. Any brine inclusions which reached the surface of a waste storage hole and subsequently became resealed might capture radioactive gases. Those inclusions would then migrate down the thermal gradient and disperse the radioactive gas.[6] However, as distance from the heat source increased, the rate of vapor-fluid inclusion migration would slow. Most such inclusions would become trapped by salt crystal boundaries. It is expected that all radioactive materials within brine inclusions would stay well within the repository formation for the hazardous life of the material.

At temperatures greater than about 250°C (480°F), the pressure caused by the thermal expansion of brine becomes so great as to cause an "explosion" in unconfined salt.[7] To avoid this hazard in Project Salt Vault, no more than 1% of the salt in a unit cell around a waste canister was allowed to exceed a temperature of 250°C. The unit cell is defined as a symmetrical unit around an in-place canister whose upper and lower boundaries are the planes of the canister ends and whose radius is half the distance to the center of the nearest canister.

Some minerals other than salt also react by releasing water at high temperatures. Among these are some clays found in shales and mudstones and several hydrated saline minerals that may exist in evaporite sequences. Gypsum is one of the most abundant and troublesome of these. It dehydrates at atmospheric pressure at temperatures between 110° and 200°C (230° and 390°F). A cubic meter (35.3 $ft^3$) of gypsum may produce as much as 0.48 $m^3$ (17.0 $ft^3$) of water.[1] The rate of dewatering and the mechanisms and pathways by which freed water might escape or be recombined would have to be evaluated for each host formation.

The effects of high temperatures on the strength of the repository rocks would be of special concern during emplacement and retrieval of wastes. Problems could result from the tendencies of plastic rocks to creep and of crystalline rocks to fracture and spall when heated. Shales might lose strength if they dehydrated, and floor heave and the fall of roof rock might increase with temperature. To limit room closure during Project Salt Vault, no more than 25% of the salt in a unit cell was allowed to exceed 200°C (390°F).[8] This limit was somewhat site-specific because it was based on the overburden pressure of the Lyons site [about 300 m (1000 ft) of sediments]. This criterion has no validity for other rock types.[9]

The validity of existing guidelines for far-field temperature increases is difficult to substantiate. The guidelines are in part a response to environmental concerns, and in part merely a specification of levels of temperature increase that Project Salt Vault did not exceed:[10]

1. The temperature increase at the ground surface directly above the buried wastes is less than 0.6°C (1°F).[8]

2. The temperature increase 1500 m (4900 ft) horizontally from the burial section of the repository is less than 0.6°C (1°F).[8]

3. Aquifers at depths less than 30 m (10 ft) do not increase in temperature more than 8°C (14°F).[10]

4. Aquifers at depths of 90 m (295 ft) do not increase in temperature more than 38°C (68°F).[10]

The surface temperature criteria are inadequate because flora and other organisms are also dependent upon subsurface temperatures, and the limiting increases probably differ with geographical regions. The horizontal temperature rise limit has yet to be justified. It is almost inconceivable that a repository that met other design criteria would not meet this one. The limits on temperature increases in aquifers should not be set arbitrarily as those given above, but need to be based on ecological studies. However, in the absence of more rigorously determined criteria, the above limits on temperature increase at a repository are still the guidelines.

Potential repository rock must be tested to determine its thermal properties and its thermomechanical and thermochemical peculiarities. In general, the greater a rock's ability to dissipate heat, the more suited it would be as host for a repository for nonretrievable storage of nuclear waste.

### D.2.2 Radiation Effects

The rock adjacent to a waste canister would be exposed to extensive radiation. It is possible that radiation energy might be stored in this rock because of the crystal lattice damage by gamma irradiation. A sudden increase in temperature might release the stored energy as excessive heat and mechanical energy.[1] Few details of radiation energy storage are known for any rocks except salt. In salt formations, thermal annealing at temperatures above 150°C (300°F) limits the storage of energy in the salt exposed most directly to radiation. Consequently, radiation energy storage is not considered as a major problem in salt.

Radiation effects also include the creation of gaseous effluents by radiolysis. Important radiolysis products from a salt repository would include $H_2$, $O_2$, and possibly $ClO_3^-$ and $BrO_3^-$.[11] If present, $Mg(BrO_3)_2$ might give off some $Br_2$. Many of these reactions would occur within the brine inclusions.[5] Hydrolysis of $MgCl_2$, present in some brines, would produce HCl, which would

increase corrosion of the canisters. Corrosion reactions between the metal canister and water vapor might also produce large quantities of $H_2$. Hydrogen explosions would be unlikely unless the storage hole plug were tight enough to permit large pressures of hydrogen to develop in the waste hole in the presence of sufficient oxygen. A hydrogen explosion in an abandoned storage room would not be a serious accident.[11]

### D.2.3 Faulting

Displacement resulting from faulting through a waste repository could pose a serious threat to containment if, as a result of the displacement, the wastes were exposed to the surface or to an aquifer. However, for displacement to be of consequence, the dip slip of the fault would have to equal the distance between the repository and aquifer, or surface, and the movement would have to be completed during the period that radiation of the wastes still posed a threat. It is more likely that the faulting would breach waste isolation by creating a permeable zone that could expose the repository to leachants.

Most major faults occur along the boundaries of crustal plates. Known vertical offsets on these faults are as large as 15 m (49 ft) (the Great Alaskan Earthquake of 1899), but averaged over a long time period, displacements usually have a small annual average offset. Uplifts of a few millimeters per year are the maximum for stable plate interiors. The average value is more on the order of 0.1 mm ($0.4 \times 10^{-3}$ inch) per year for a 100,000-year time reference.[3] Thus, it is suggested that most fault activity could be avoided by positioning the repository on the interior of a tectonic plate.

Tectonic faulting is always accompanied by seismic activity. Extensive maps have been developed which indicate the frequency of measured seismic activity (see Fig. D-1). This information could be used to help select a stable repository site. Regions of tectonic stress which may develop folds or faults must also be avoided. Regions having dips of a few degrees or less are the most likely to be stable.

The repository might induce stresses on the local rock structures. Differential thermal expansion could cause localized faulting when rocks were rapidly heated. The bulging or subsidence which might appear at the surface is more likely to be a gradual plastic deformation. The maximum surface displacement would be a function of the heat generated, the thermal properties of the rocks, and the depth of the repository.

Any fault which connected an aquifer with a repository might increase the hazard of the wastes' being leached; however, little damage would accrue unless the water were permitted to circulate. One such situation would occur if a fault connecting an upper and lower aquifer passed through a repository. Downward flow could leach the wastes and contaminate the lower aquifer. If the repository were in a salt formation, dissolution would eventually expose more wastes to leaching and increase the contamination rate. Although contamination of a deep aquifer is undesirable, it might be inconsequential. Normal flow velocities may be only a few kilometers a year in deep aquifers, so only by drilling in or near the buffer zone could the contamination be exposed. A much more dangerous situation would occur if the flow were upward and an overlying aquifer were contaminated. Potentiometric heads in deep aquifers that are capable of causing upward flow through a fault zone to the level of an upper aquifer are not common.[1] Should materials leached from a repository reach a shallow aquifer, the potential for widespread contamination and ingestion by plants and animals would be great. Fortunately, the downward flow condition is more likely to occur.

The volume of water that can pass downward through a fault zone depends in part on fault-zone permeability, but even more so on the availability of water. Potential recharge then becomes a factor. An increase in availability of water due to climate change, flooding, or other causes could contribute to the decrease in repository containment, especially if the host formation were soluble, like salt. When flow through a fault in a salt or shale formation is intermittent or very slight, the rock may recrystallize and heal, thus preventing additional water circulation.

### D.2.4 Volcanism

A repository subjected to volcanic activity would not only exceed allowable temperature ranges, but its containment would be breached to the extent that radioactive wastes might be liberated with hot gases, shards, or molten material. Presently active volcanic areas are easy to avoid. They may be identified by the presence of young volcanic rocks or by abnormally high geothermal gradients accompanied by seismic and tilt activities. Volcanism can usually be avoided by considering only sites on the interior of continental plates. Only 3% of all historically active volcanoes are in midcontinental areas.[3] Although the rise of magma is accompanied by extensive

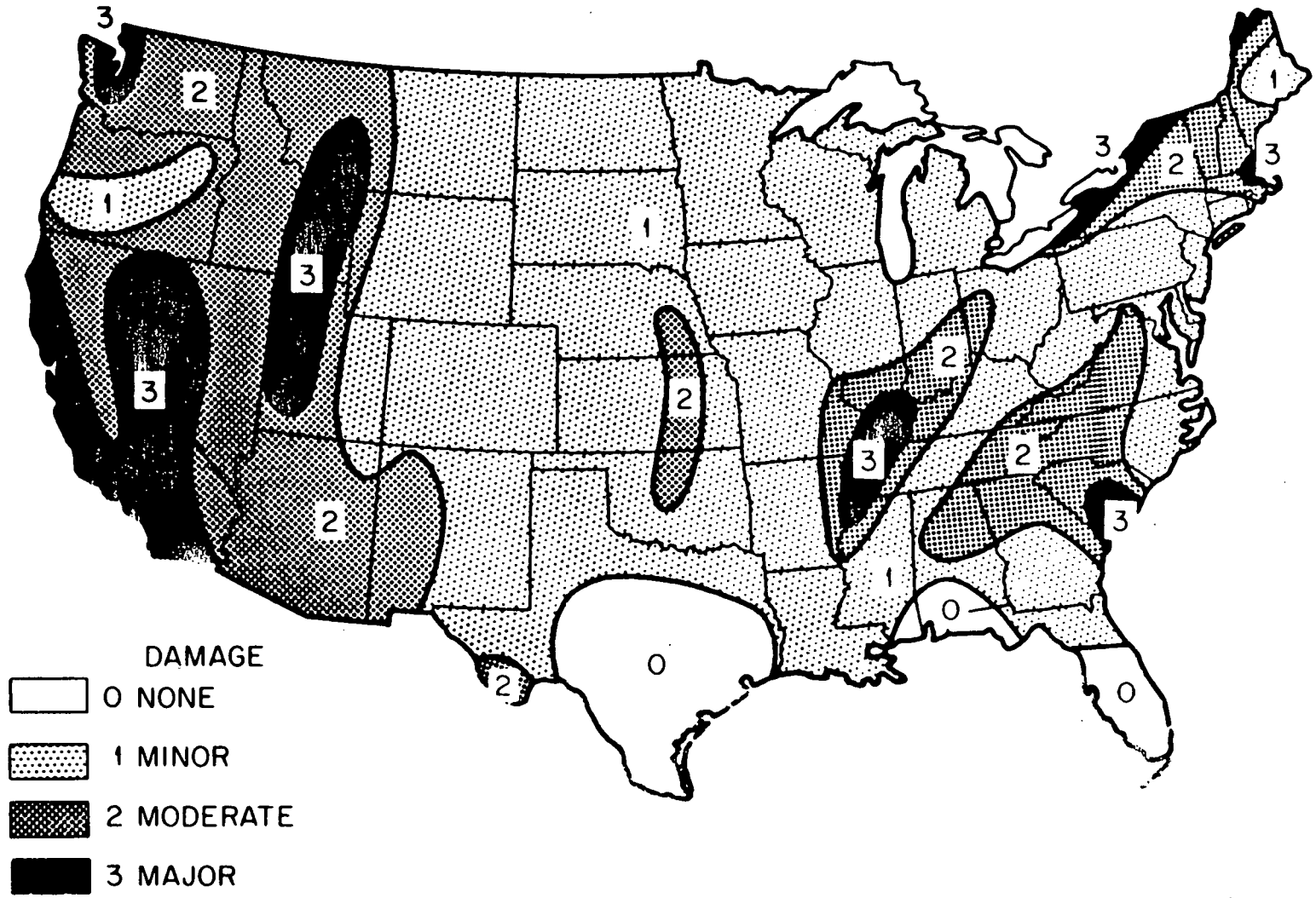


Fig. D.1. Seismic Risk Map of the United States. (Fig. II-4 in "Final Environmental Statement, Management of Intermediate-Level Radioactive Waste," U.S. Energy Research and Development Administration, ERDA-1553, September 1977.)

faulting, many fault systems have no volcanic connections; thus, the probability of a repository site being subsequently subjected to volcanism is significantly lower than the probability of massive faulting.

### D.2.5 Erosion

The fact that a shallow repository may be exhumed by erosion was mentioned previously when the minimum repository formation depth was suggested to be no less than 250 m (820 ft). Any agent--water, wind, or ice--may be responsible for removing overburden. The susceptibility of an area to erosion depends upon the surface relief, vegetation, and climate, and might be subject to change during the period that repository wastes were hazardous.

### D.2.6 Cratering

A crater is the depression produced by the explosive impact of a meteorite, bomb, or other device that shatters the earth's surface and causes fragments to become temporarily airborne. The depth of a crater extends from the earth's surface to the bottom of the breccia that resettles in the hole. Should an impact occur at a repository, the seriousness of the event would depend upon the depth of the crater. If the crater reached or exceeded the depth of waste burial, it is conceivable that some of the radioactive material could become airborne. If only the fracture zone reached the burial site, the overburden would be shattered and groundwater could reach the protective formation, and possibly the wastes, but no instantaneous release of radionuclides would occur.[3]

Geographic areas prone to meteorite impact are difficult to identify. At the current rate of bombardment, the probability of a repository's being penetrated by a meteor is small, regardless of geographical area. The best protection is the depth of burial.

Impact by nuclear weapons may not be geographically random, and sufficient repository depth may be the only way to ensure waste isolation. Nuclear weapons currently are of such a size that a surface burst would not penetrate a sealed repository that is deeper than 500 m (1640 ft). The largest deployed missile is reported to be capable of carrying a 25-megaton warhead. A surface burst of this magnitude would generate a 270-m (885-ft) deep crater with a fracture zone down to about 400 m (1312 ft) in a geological material with the physical properties of dry soil. The crater would be somewhat smaller in salt. For a 50-megaton weapon, the potential crater depth in dry soil would increase to only 340 m (1115 ft) and the fracture zone to 500 m (1640 ft). Thus, a nuclear explosion might be a hazard for more shallow repositories and might be considered as a limiting factor for depth selection.[3]

### D.2.7 Drilling and Mining

A stipulation that no repository be located in or near a mineral deposit of potential economic value would minimize the chance that stored wastes would be exposed by subsequent drilling or mining. The surface above a storage site would be marked with monuments that identified the repository and delineated a buffer zone. Even if the site was drilled to the depth of the repository, the chance of hitting within 50 cm (20 inches) of a canister would be small.[3] Surface contamination resulting from such drilling would probably be minimal and localized.

Another potential problem related to drilling is the possibility of failure of borehole plugs. Wells drilled during exploratory phases of repository construction would have to be plugged in the best available manner. Special problems with dissolution might exist if a borehole passing through a potential salt repository formation intercepted both overlying and underlying aquifers. This problem has been described in greater detail in connection with fault hazards (Sec. D.2.3).

## D.3 SECONDARY PROTECTION MECHANISMS

Selection of candidate formations for use as geological repositories for nuclear wastes would involve making estimates of the potential for and consequences of radionuclide migration. Release of radionuclides from a repository could result from two general types of events or processes: (1) catastrophic events, like meteorite impact, and (2) degradation processes, like erosion of the protective formation. Analyses of the risks and consequences of these two types of events are somewhat different. Catastrophic events are usually assessed by the type and probability of the initiating event. For degradation processes, phenomena which can release radioactivity are first assumed to occur, then emphasis is placed on analyzing the rates and characteristics of resulting radionuclide release and migration.

Except with drilling or explosive breach of a repository, the primary agent that would transport radionuclides is water. Evaluation of radionuclide migration thus presumes that water enters the repository, acts to release the radioactivity (e.g., by leaching), and transports the radioactivity through the surrounding media to the biosphere. An evaluation of radionuclide release and migration from a repository intruded by water involves analysis of (1) the release of the radioactivity from the waste material to the water and (2) the subsequent movement through the surrounding media. Waste forms and containers can be designed to impede leaching, but ultimately they would be breached, and the movement of the water containing radioactivity would become all-important.

Some geologic media surrounding a repository could impede radionuclide migration by sorption. Sorption includes such phenomena as adsorption, ion exchange, colloid filtration, reversible precipitation, and irreversible mineralization. The result of these mechanisms is that nuclides move at a lower (often much lower) velocity than water through most media.[1] A few rocks, such as salt, have no sorptive capabilities and allow radionuclides to move at the same rate as the water.

Some radionuclides, such as the isotopes of uranium and thorium, are strongly sorbed by media that have sorption capability; others, such as strontium and neptunium, are moderately sorbed. A few, such as technetium and iodine, are poorly sorbed by most geologic media.[1] The sorption of the isotopes of a particular element is expressed as a sorption equilibrium constant, K, that is a ratio of water velocity to nuclide migration velocity. For particulate media, $K = 1 + K_d\rho/\varepsilon$; where $\rho$ is the bulk density and $\varepsilon$ is the void ratio. $K_d$ is a measure of the moles of the radionuclide in the sorbed state per unit mass of the geologic medium divided by the moles of radionuclide in the dissolved state per unit volume of groundwater when the groundwater and medium are in equilibrium.[1]

For faulted media, $K = 1 + K_a R_f$. $R_f$ is the surface-to-volume ratio of the fault. $K_a$ for faulted media is similar to $K_d$ for particulate media.

The $K_d$ and $K_a$ values are based on several parameters, including the pH of the water, concentration of dissolved salts (such as NaCl), solution temperature, and sometimes nuclide concentrations. Although laboratory modeling that resembles actual conditions is difficult to achieve, predictions of radionuclide discharge rates to the biosphere have been made using approximate K values. An example of such modeling is shown in Figure D-2.

Predictions for radionuclide migration from a particular repository must be based upon sorption measurements. All such measurements in the past have been conducted on near-surface materials. Based on the tests conducted to date, the ions of greatest concern are Tc-99, I-129, Ra-226, U-234, Pu-238, and Np-237.

The second important secondary protection mechanism is the long path that should exist between stored wastes and usable groundwater. Distance enhances sorptive characteristics and diffusion, and lessens the chance of the radionuclides ever reaching the groundwater. In the case of an aquifer overlying a repository, head differential would be another advantage of distance.

## D.4 REQUIREMENTS FOR RETRIEVABILITY

In addition to all the requirements for permanent isolation of wastes, a geologic formation containing a repository for the retrievable storage of nuclear wastes would have to be of such a nature as to permit the safe removal of wastes for a period up to 25 years. The ideal host rock would maintain its originally mined dimensions and mechanical characteristics. The effects of mine depth and waste/rock interactions would be insignificant. Ideally, host rock for a retrievable storage repository:

- would not be prone to accelerated creep or flow at high temperatures;
- would not expand so much when it was heated as to cause the stored wastes to be squeezed and frozen in place;
- would not be prone to heaving, spalling, or rock burst when exposed to high temperatures;
- would readily dissipate the heat generated by the stored wastes;
- would provide a radiation shield and at the same time not store large amounts of radiation energy; and
- would not accelerate the corrosion of waste containers.

Characteristics of the PNL Geosphere Migration Model

1. All of the Year 2000 U.S. nuclear power economy waste is contained in a nonsalt repository surrounded by a western U.S. desert geologic medium.
2. The waste is contacted by groundwater from a typical U.S. desert starting at varying times between the Year 2000 and the Year 10,002,000.
3. The waste is leached by that groundwater at varying rates between 0.00003 and 100%/year.
4. The groundwater moves from the repository through a one-dimensional column of the medium and discharges into a surface water body.
5. The sorption equilibrium constants are based on measurements and estimations for U.S. desert subsoils.
6. The groundwater velocity is 1 ft/day.
7. The path length from the repository to the surface water body varies from 0 to 100 miles.
8. The axial dispersion coefficient is 0.008 $cm^2/min$.

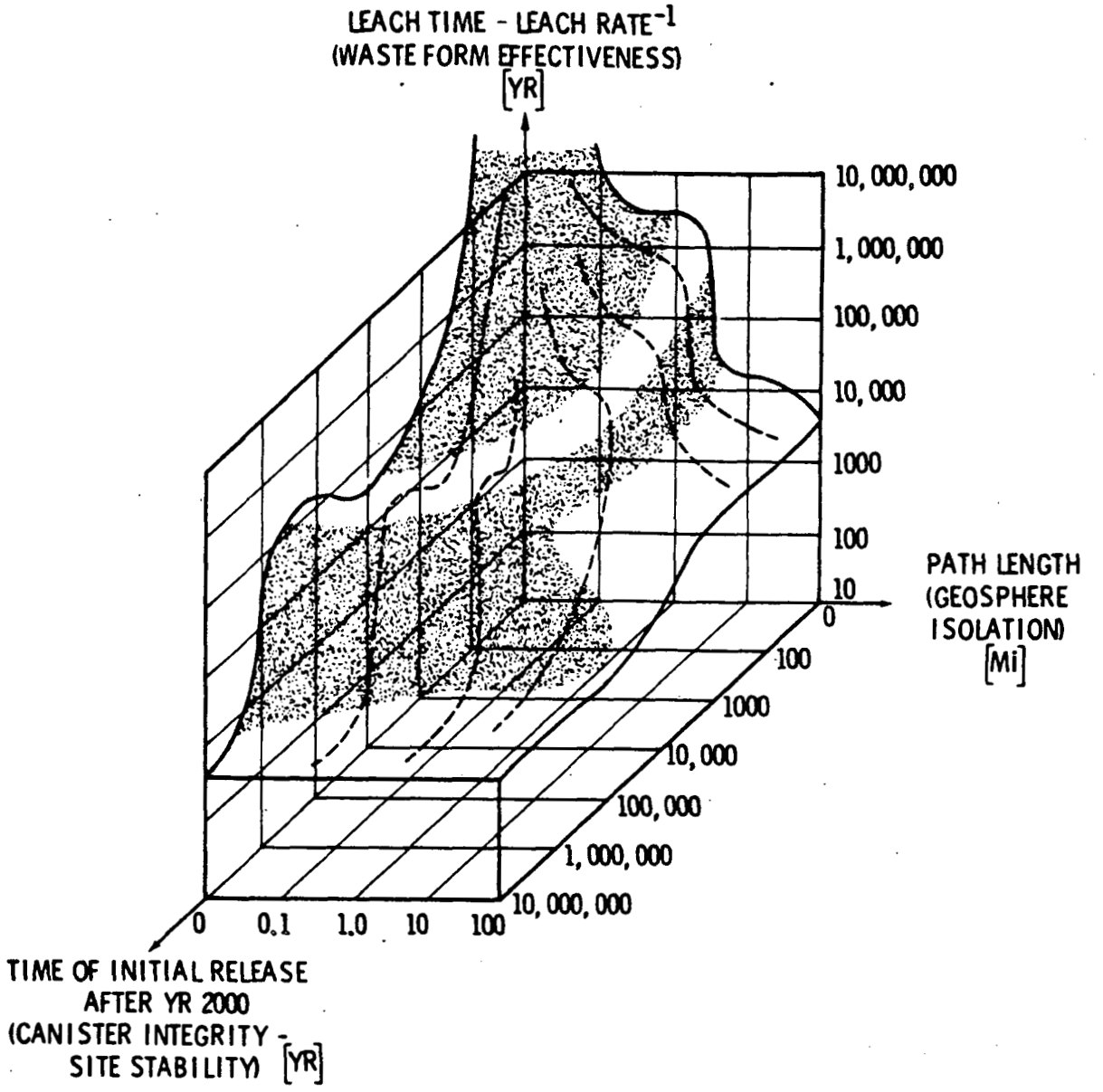


Fig. D.2. Waste Management Control Surface for Incremental Background Dose with No Partitioning. (Adapted from "Alternatives for Managing Wastes from Reactors and Post-Fission Operations in the LWR Fuel Cycle," U.S. Energy Research & Development Administration, ERDA 76-43, May 1976.

No known formation meets all these criteria. Salt has a relatively high thermal conductivity and, consequently, would be subjected to lower temperatures. Its properties, however, are sensitive to temperature. Flow is a major problem. Flow which squeezed the waste canisters might be combated with thick, mild-steel sleeves, but no suitable solution has been discovered for flow which causes room closure. This problem alone might be severe enough to exclude salt formations from consideration for a 25-year retrievable repository.

Granite and basalt are under serious consideration for long-term retrievable repositories because of their structural competence at high temperatures. They are, however, subject to fracturing, spalling, and rock burst at extreme temperatures. This might make protection of workers, waste transporters, and stored wastes difficult and costly. An igneous rock repository would have to be larger than a salt repository because of lower thermal conductivities. Igneous rock may also be more costly to mine. However, igneous rock may be necessary to meet the requirements for competence for a retrievable repository.

## D.5 SUMMARY

Salt rock has been studied carefully and is considered by many, including the National Academy of Sciences-National Research Council,[2] to be one of the best types of host rocks for a geological repository. This is because salt is impermeable, plastic, and in time would make a very tight waste container. A salt formation at least 250 m (820 ft) deep would be protected against erosion. At a depth of at least 280 m (920 ft), and especially at 350 m (1200 ft) or deeper, a salt-host repository would be protected from nuclear attack. Because of structural considerations, 1500 m (4900 ft) is probably the maximum practical depth for a repository in salt. The minimum thickness must be determined for each specific set of geologic and waste storage conditions. The formation should extend laterally for a distance great enough to maintain structural integrity.

A salt repository would be subject to dissolution if water was able to circulate through fractures, drill holes, or any other connection between enclosing aquifers. Radionuclides could be removed by leaching, but sorptive tendencies of rocks adjacent to the salt repository might retard the movement of these waste materials. The distance between a repository and an aquifer would provide additional passive protection. Most water available for leaching radionuclides would not be likely to have the potentiometric head necessary for contamination of a shallow aquifer.

The consequences of most natural catastrophies could be avoided by careful exploratory geology. In no case should a repository be built near the edge of a crustal plate or in an earthquake-prone, fault-prone, or volcanically active area. Careful attention to the thermal properties of the rock and repository design emplacement density will minimize undesirable waste/rock reactions. Depth is sufficient protection against impact of most meteorites or bombs.

Provided that care were taken to ensure that the repository was not in an area that might someday be economical to mine, inadvertent drilling and excavation should not be a potential problem. The surface above a repository and buffer zones around it should be marked with permanent markers which identify the repository and warn against drilling. Even if drilling occurred, the waste canisters are not likely to be penetrated.

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