 ANL/EIS-11

GENERIC WASTE MANAGEMENT CONCEPTS FOR SIX LWR FUEL CYCLES

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Project Leader

U of C-AUA-USDOE

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 edecessor, 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.

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*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).**

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*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.

<u>Table based on information from</u>:

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.

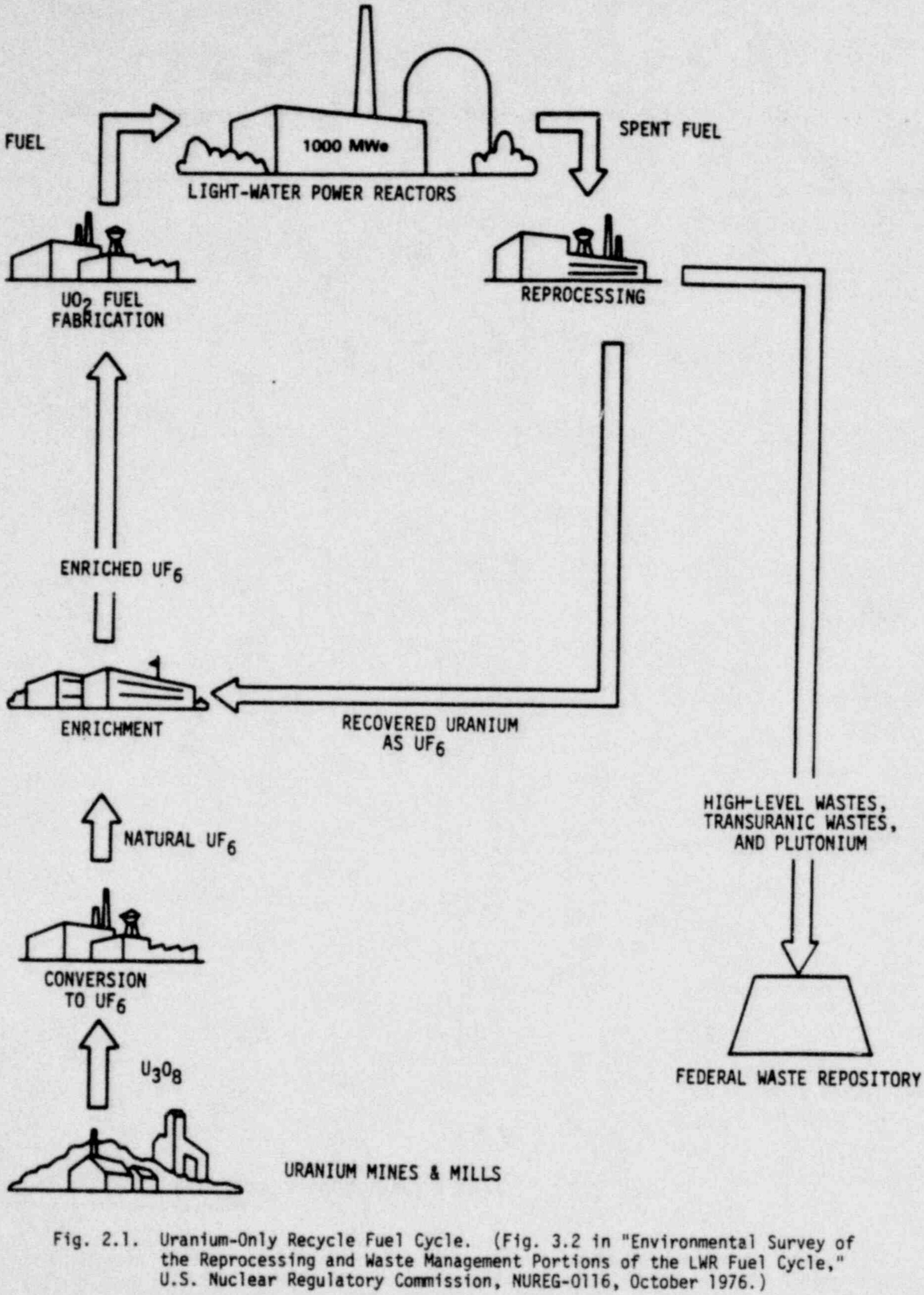


Fig. 2.1. Uranium-Only Recycle Fuel Cycle. (Fig. 3.2 in "Environmental Survey of the Reprocessing and Waste Management Portions of the LWR Fuel Cycle," U.S. Nuclear Regulatory Commission, NUREG-0116, October 1976.)

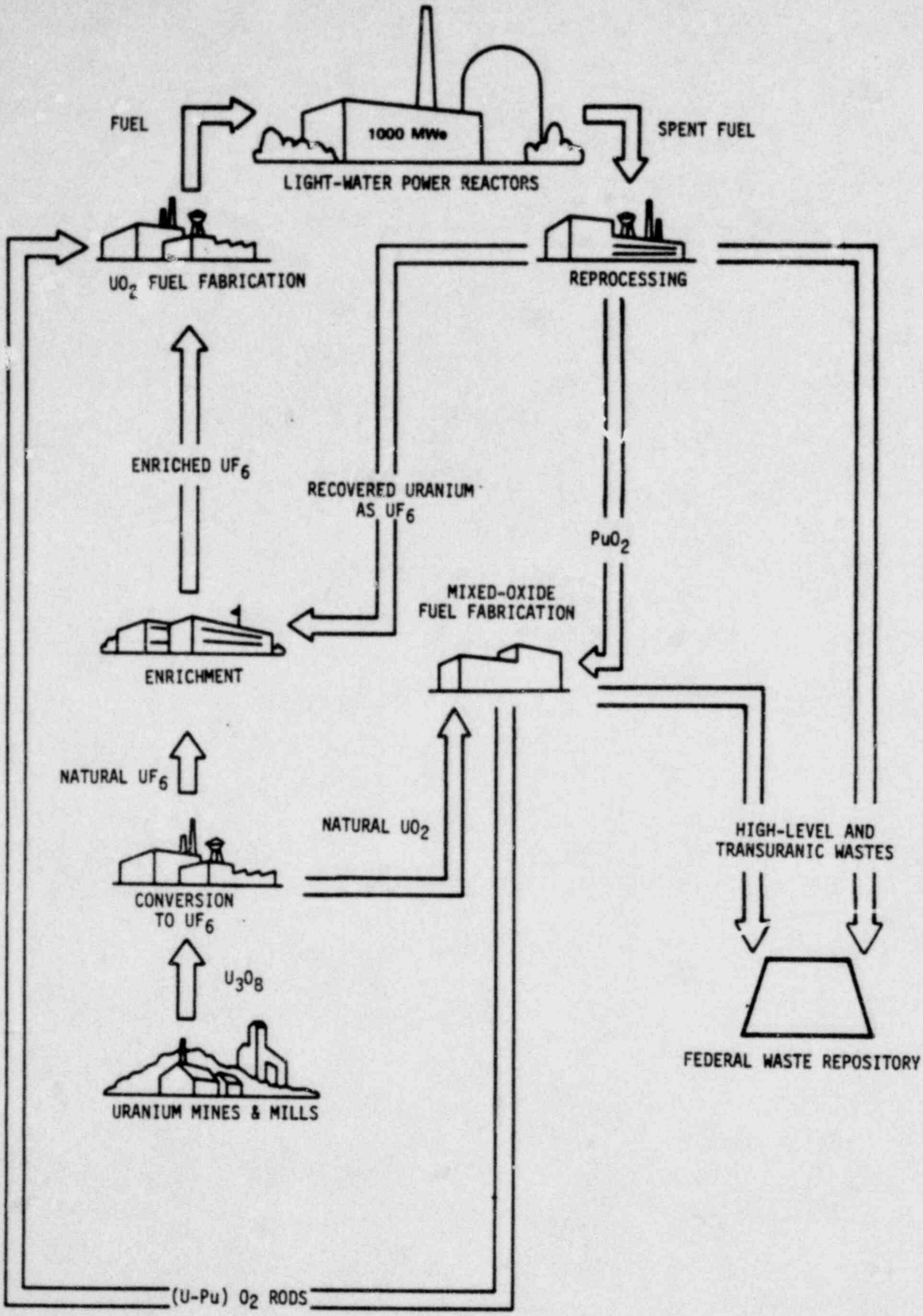


Fig. 2.2. Full Recycle Fuel Cycle. (Fig. 3.3 in "Environmental Survey of the Reprocessing and Waste Management Portions of the LWR Fuel Cycle," U.S. Nuclear Regulatory Commission, NUREG-0116, October 1976.)

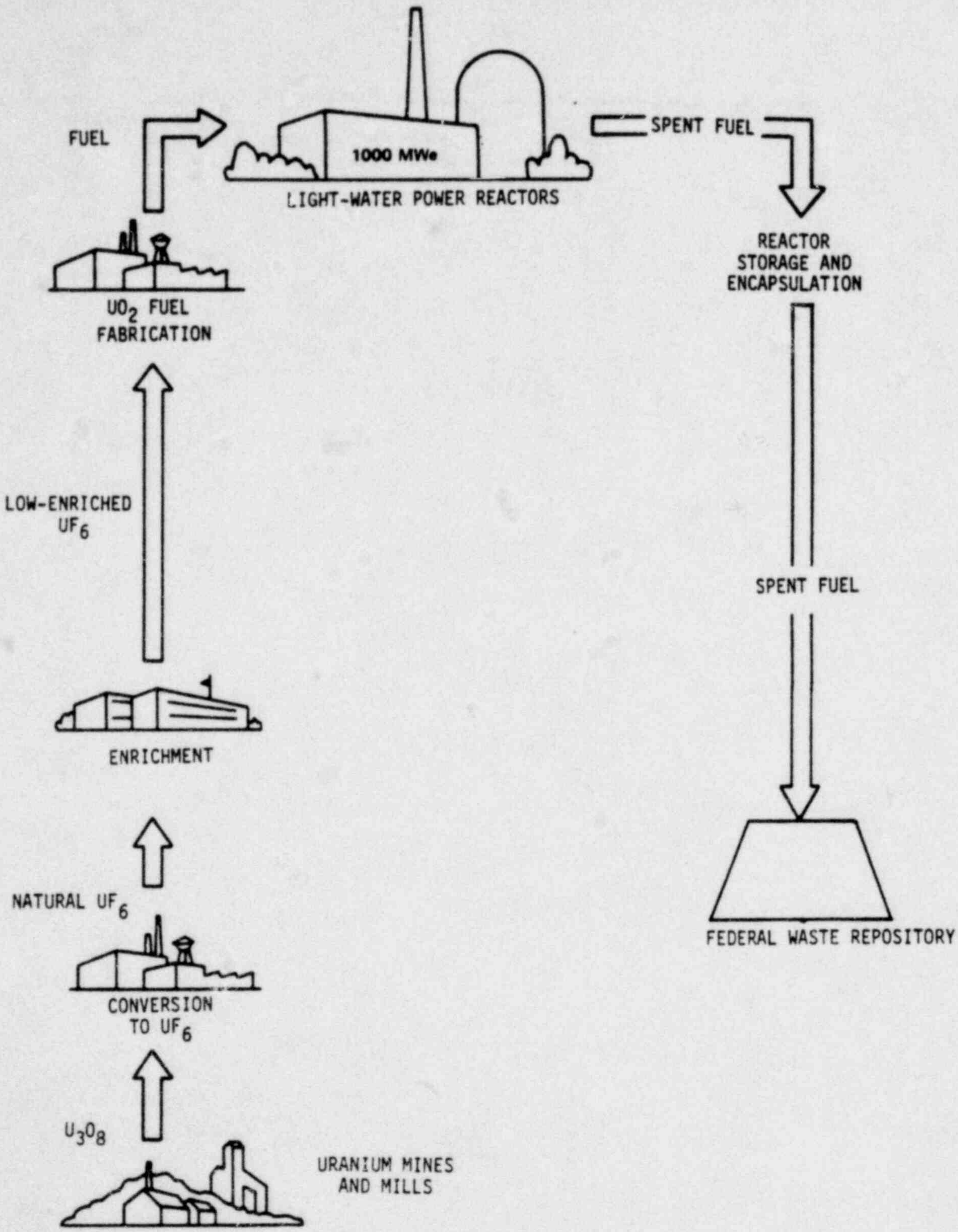
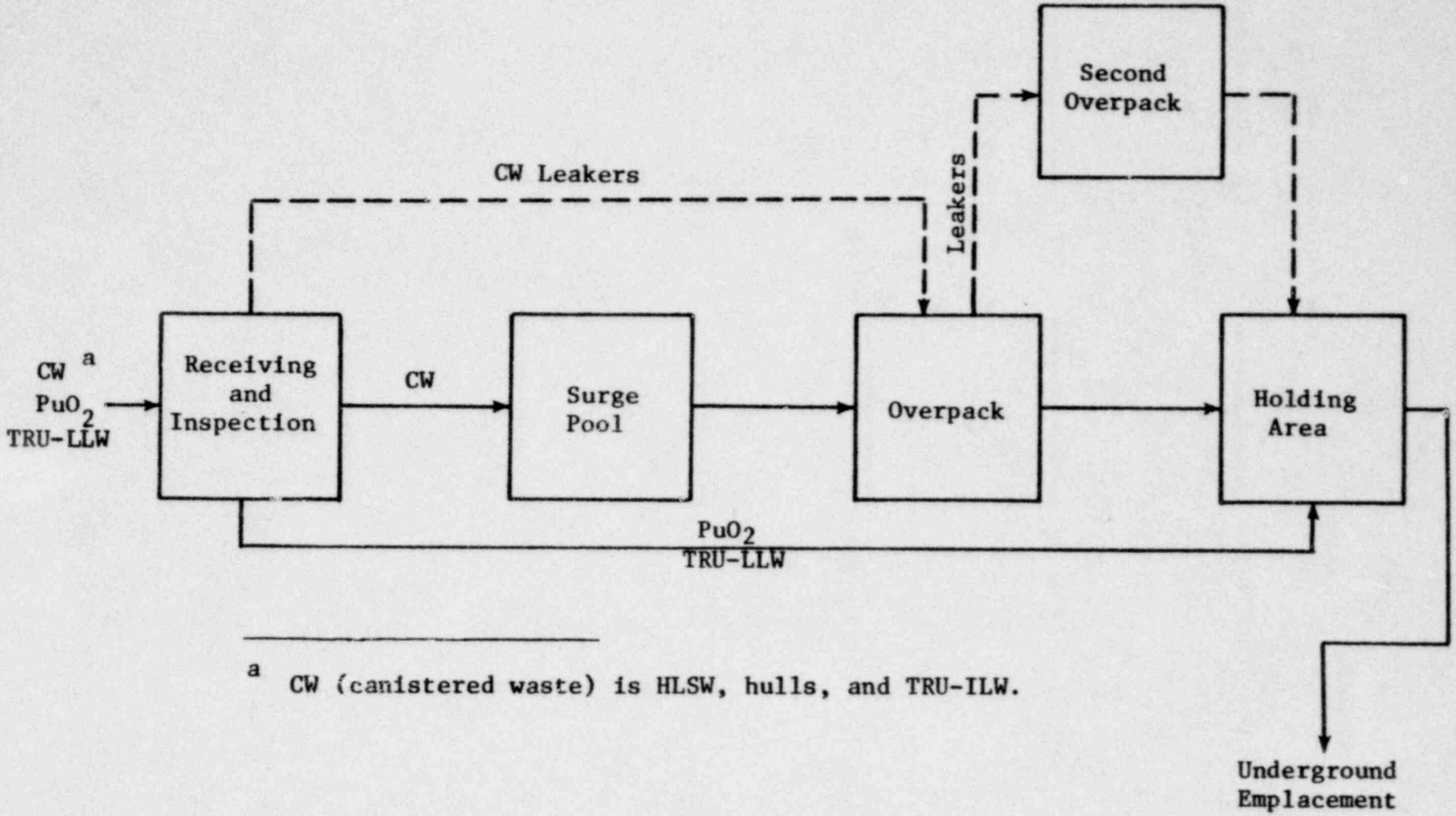


Fig. 2.3. No-Recycle Fuel Cycle. (Fig. 3.1 in "Environmental Survey of the Reprocessing and Waste Management Portions of the LWR Fuel Cycle," U.S. Nuclear Regulatory Commission, NUREG-0116, October 1976.)



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

Fig. 2.4. Flow Diagram for Reprocessing Wastes Repository.

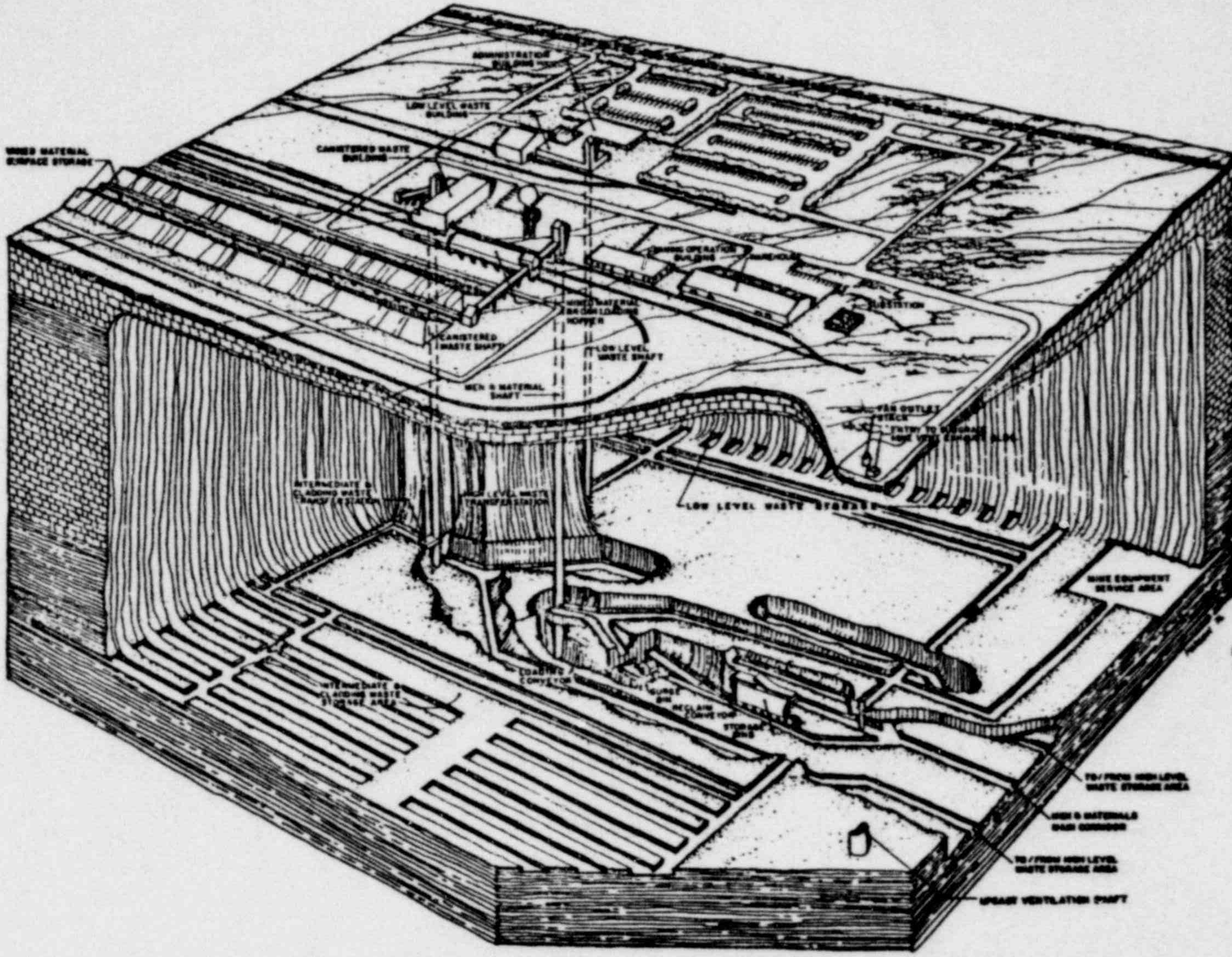


Fig. 2.5. Schematic Diagram of Waste Isolation Facility. (From "Waste Isolation Facility Description - Bedded Salt," Office of Waste Isolation, Y/OWI/SUB-76/16506, September 1976.)

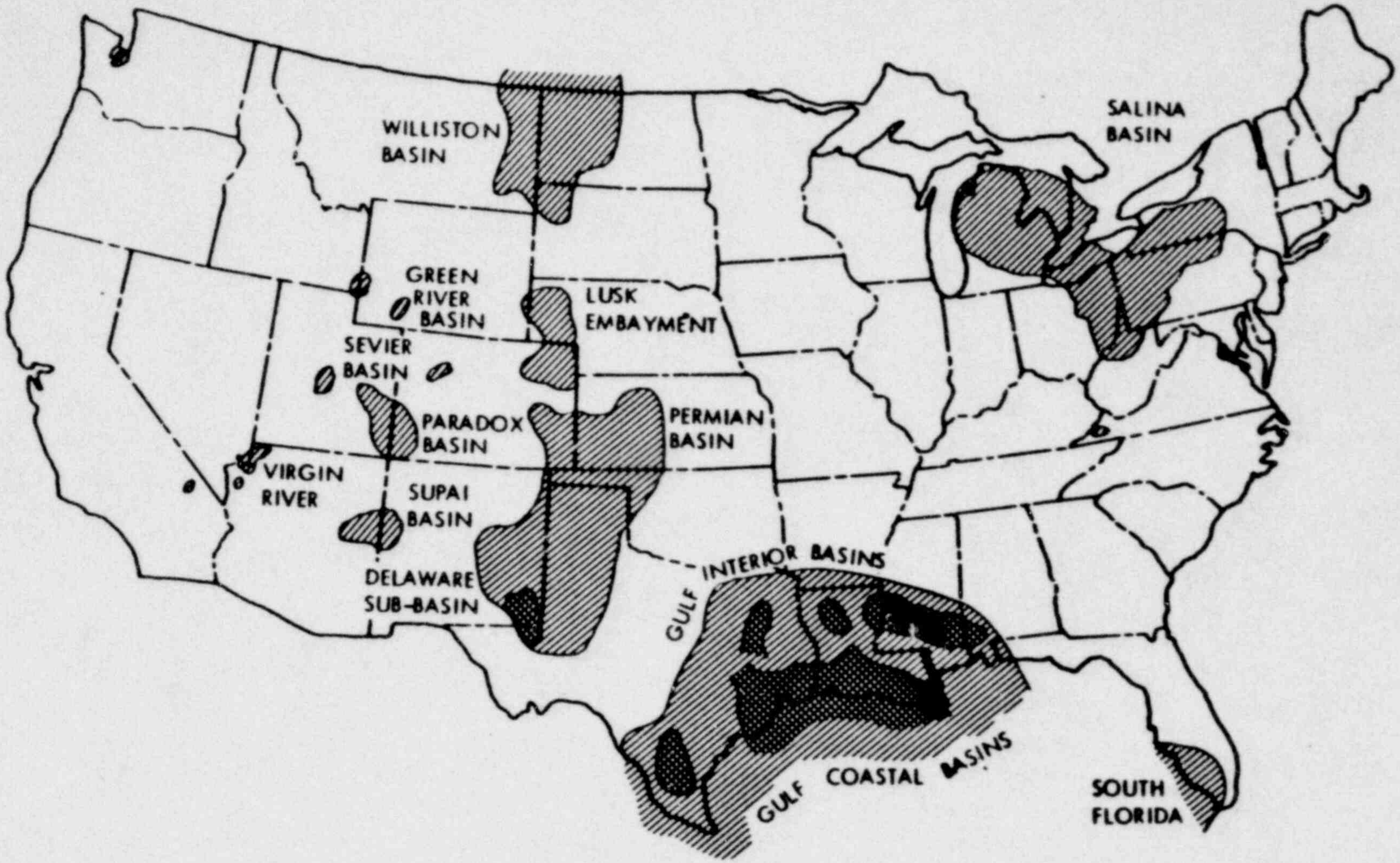


Fig. 2.6. Rock Salt Deposits in the United States. [Fig. 4 in "Proceedings of the Symposium on Waste Management, Tucson, Arizona, October 3-6, 1976," CONF-761020, R. G. Post (editor), University of Arizona, the Arizona Atomic Energy Commission and the Western Interstate Nuclear Board.]

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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.

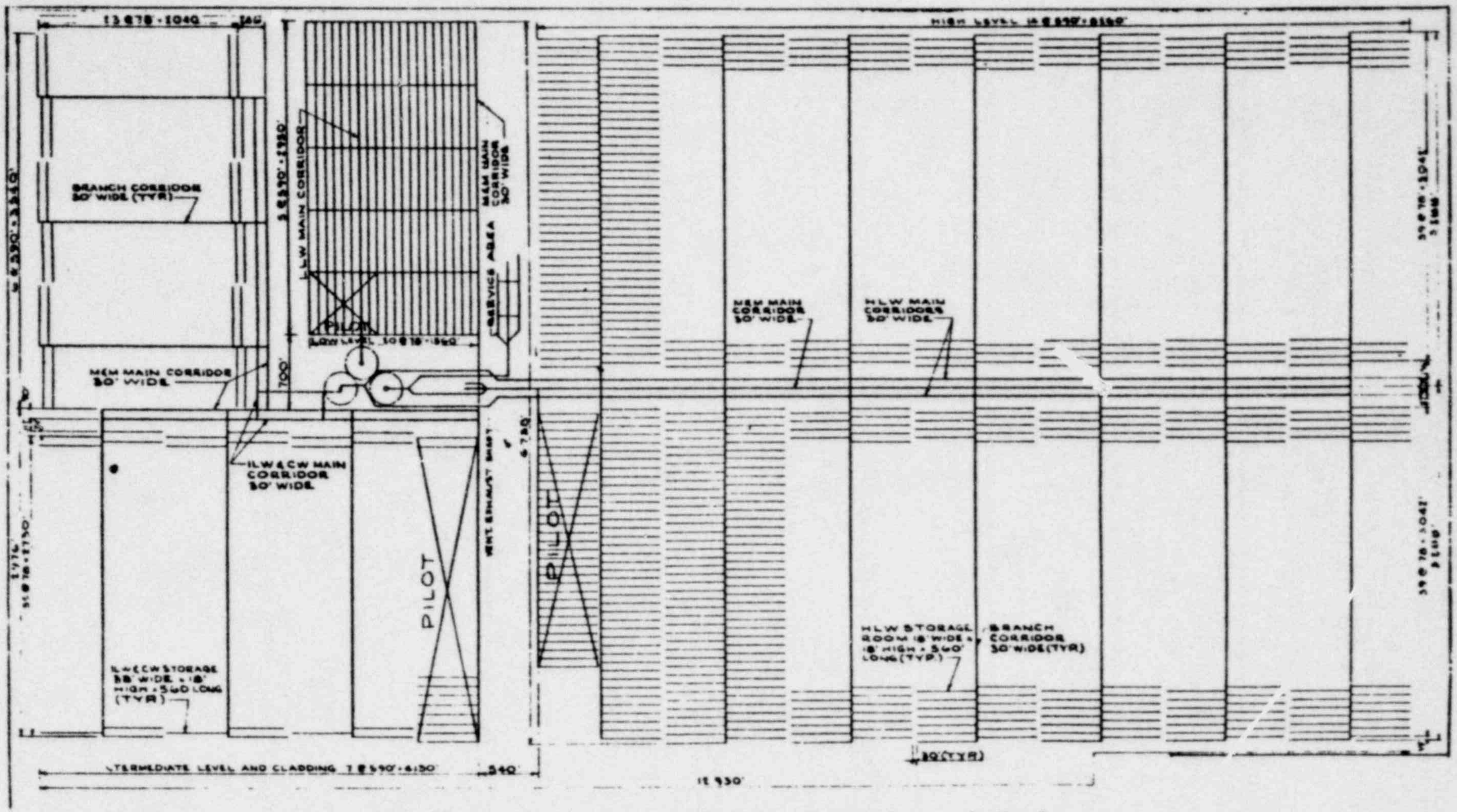
After transport through the shaft to the mine, a waste canister would pass through a mine-level receiving station before entering the storage area. The receiving station would be a shielded enclosure with a viewing gallery. A top view of the mine storage corridor and room arrangement is given in Figure 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.)

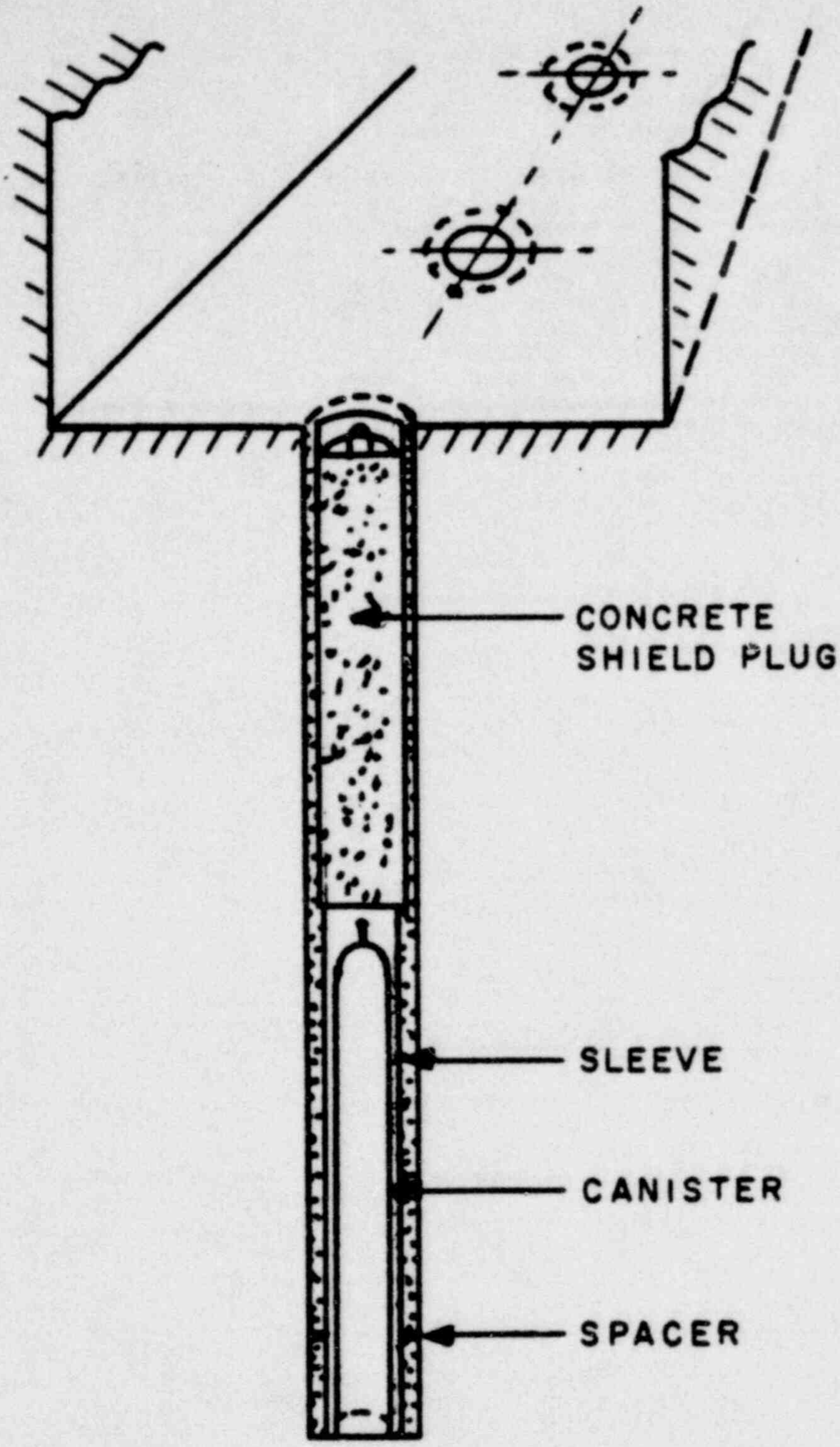


Fig. 2.8. Retrievable Emplacement. (From: "Waste Isolation Facility Description - Bedded Salt," Office of Waste Isolation, Y/OWI/SUB-76/16506, September 1976.)

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

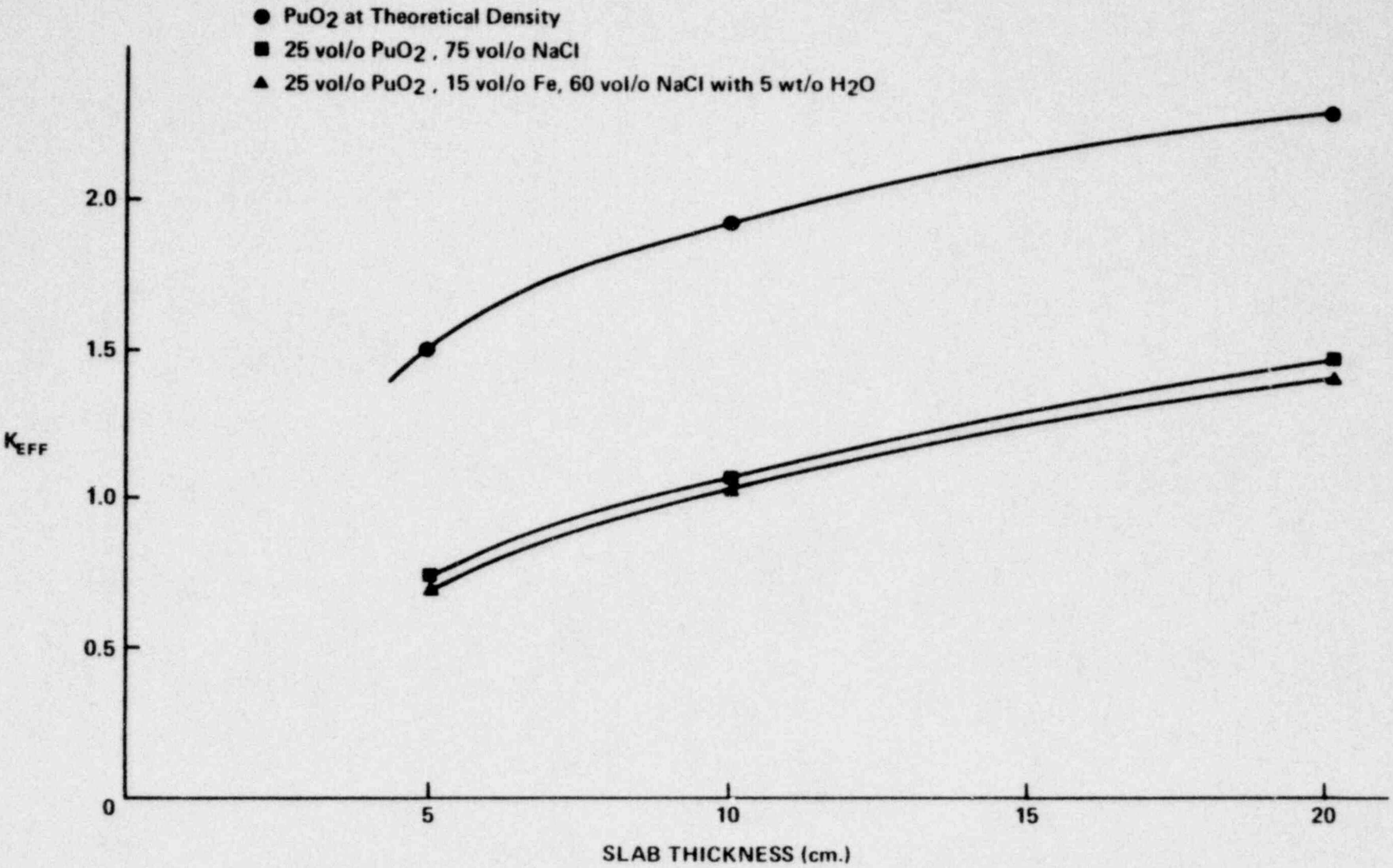


Fig. 2.9. K_{eff} for PuO_2 in a Salt Repository, Slab Geometry. [Fig. 4.12 in "Public Comments and Task Force Responses Regarding the Environmental Survey of the Reprocessing and Waste Management Portions of the LWR Fuel Cycle (NUREG-0116)," U.S. Nuclear Regulatory Commission, NUREG-0216, March 1977.]

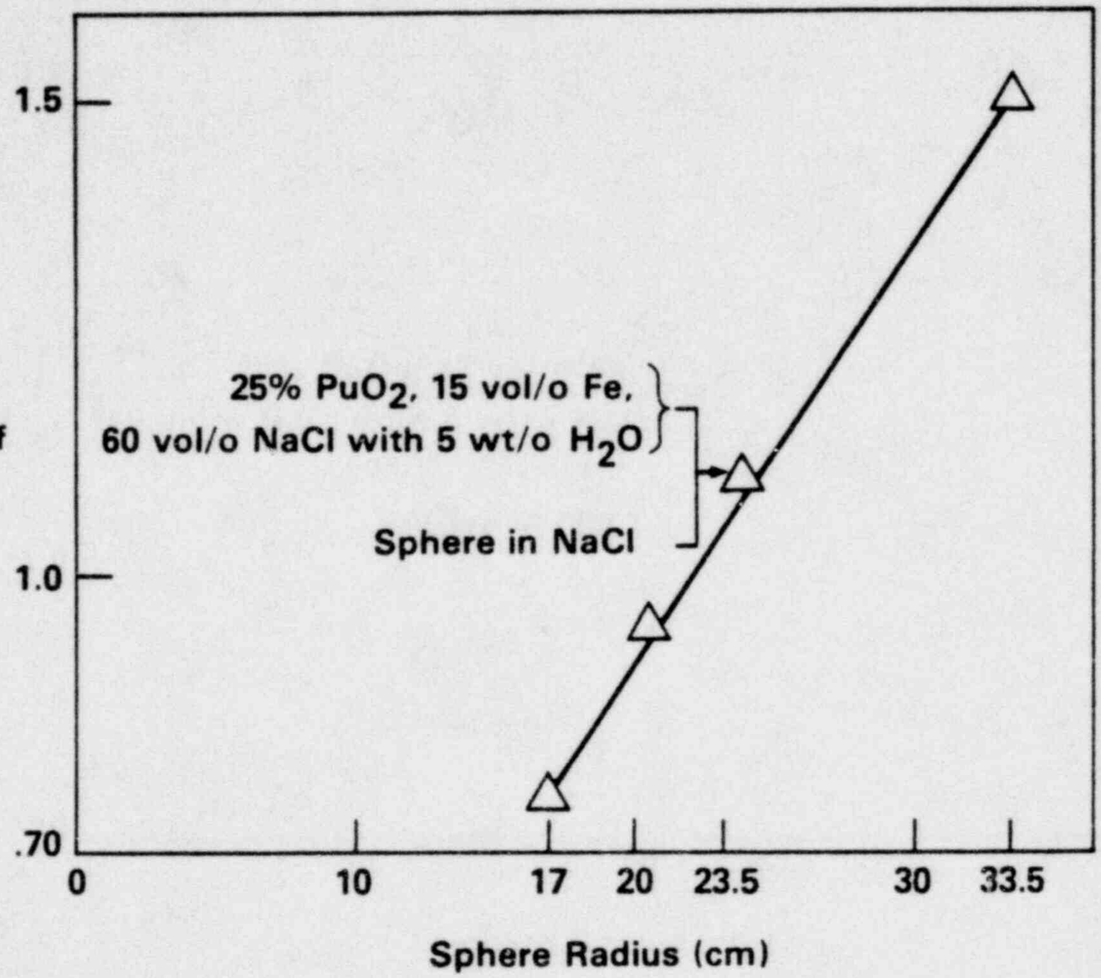


Fig. 2.10. K_{eff} for PuO_2 in a Salt Repository, Sphere Geometry. (Fig. 4.13 in Environmental Survey of the Reprocessing and Waste Management Portions of the LWR Fuel Cycle," U.S. Nuclear Regulatory Commission, NUREG-0116, October 1976.)

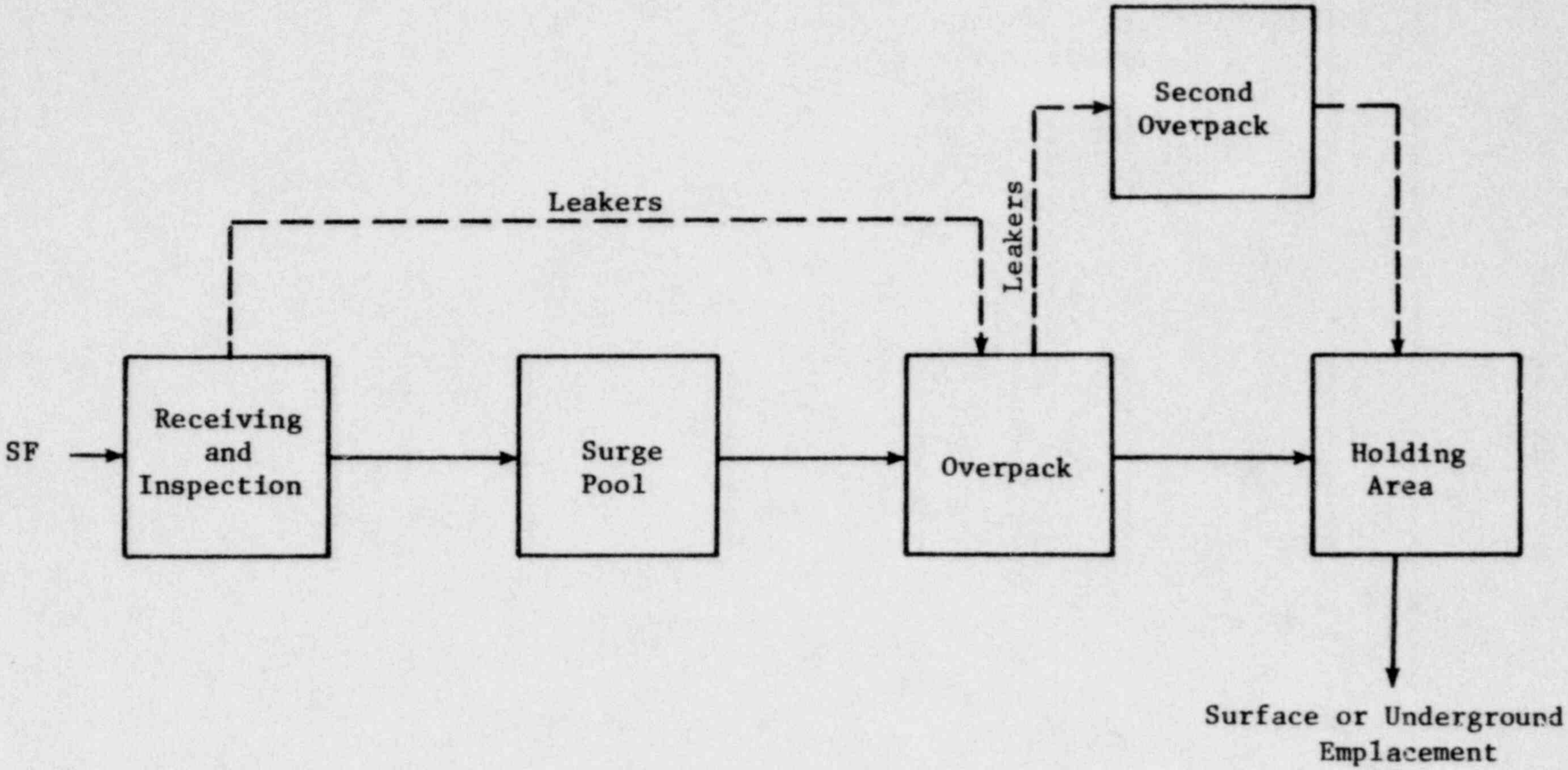


Fig. 2.11. Flow Diagram for Spent Fuel (SF) Repository.

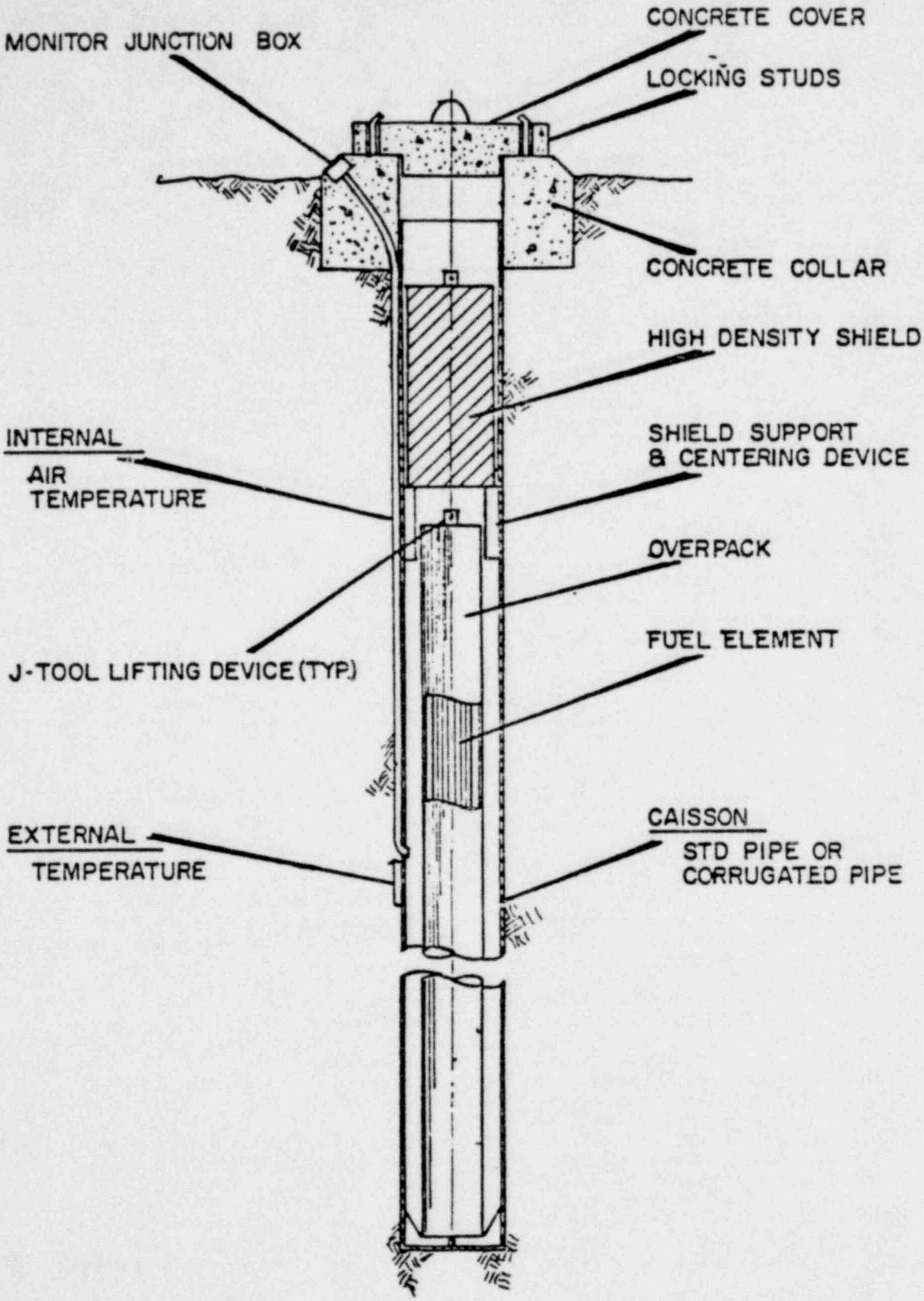


Fig. 2.12. Spent Fuel Storage Caisson.

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

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*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.

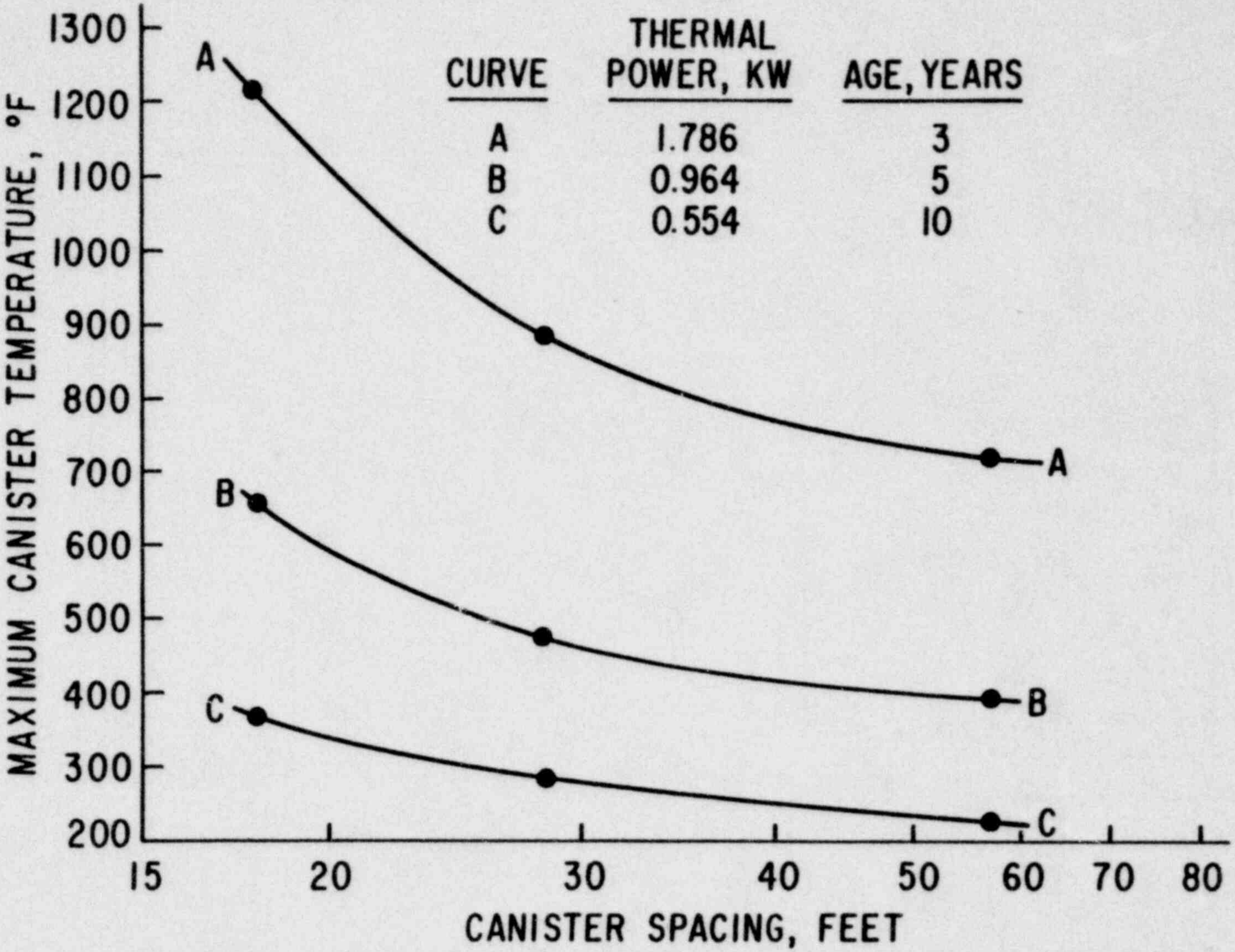

Fig. 2.13. Canister Temperature vs. Spacing for PWR Fuel (caisson storage). [From "Spent Unreprocessed Fuel Facility, Engineering Study of Storage Concepts (Draft)," prepared by Atlantic Richfield Hanford Company for U.S. Energy Research and Development Administration, 1 June 1977.]

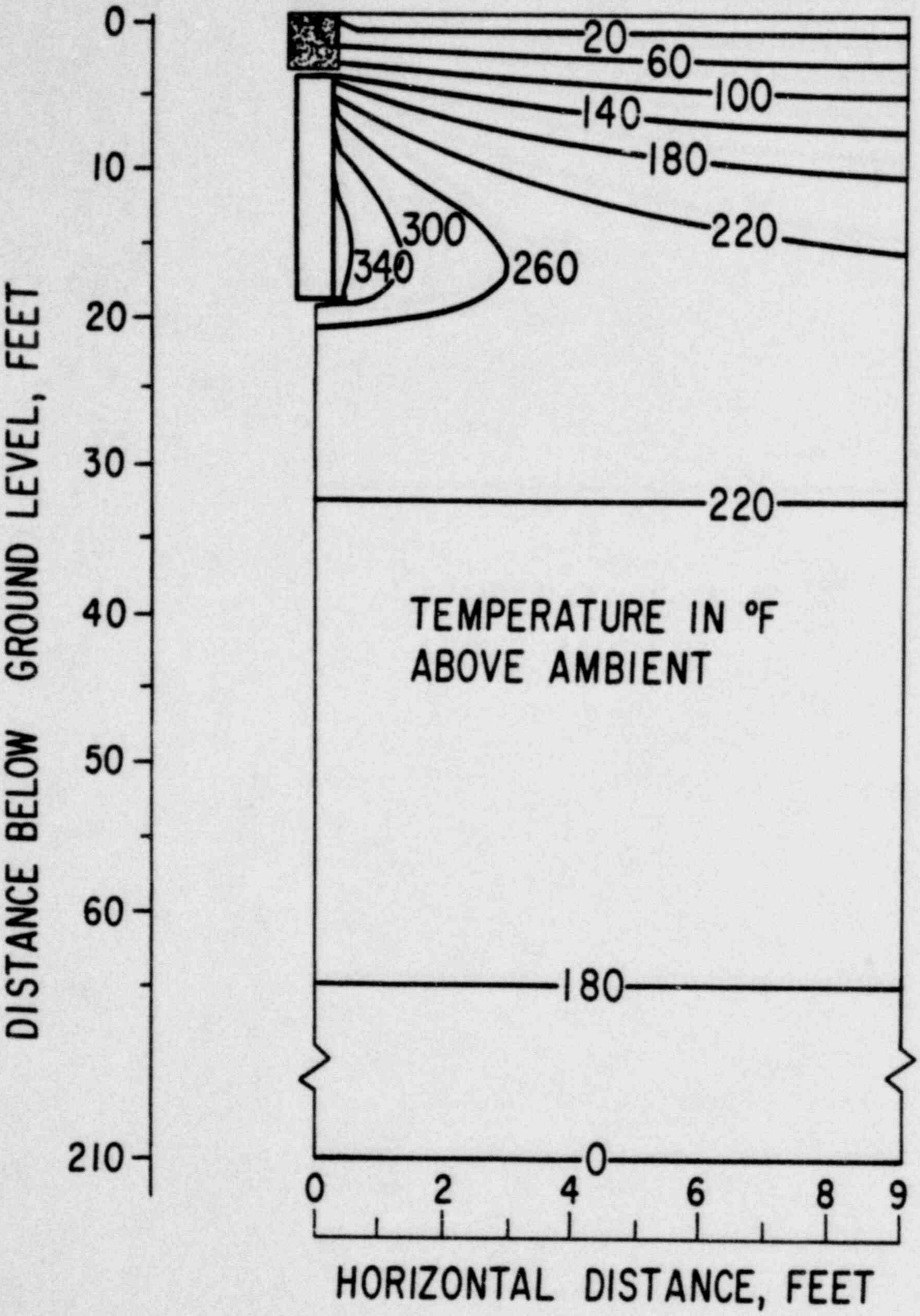

Fig. 2.14. Caisson Isotherms for PWR Fuel Ten Years after Discharge (spacing of 5.5 m). [Adapted from "Spent Unreprocessed Fuel Facility, Engineering Study of Storage Concepts (Draft)," prepared by Atlantic Richfield Hanford Company for U.S. Energy Research and Development Agency, 1 June 1977.]

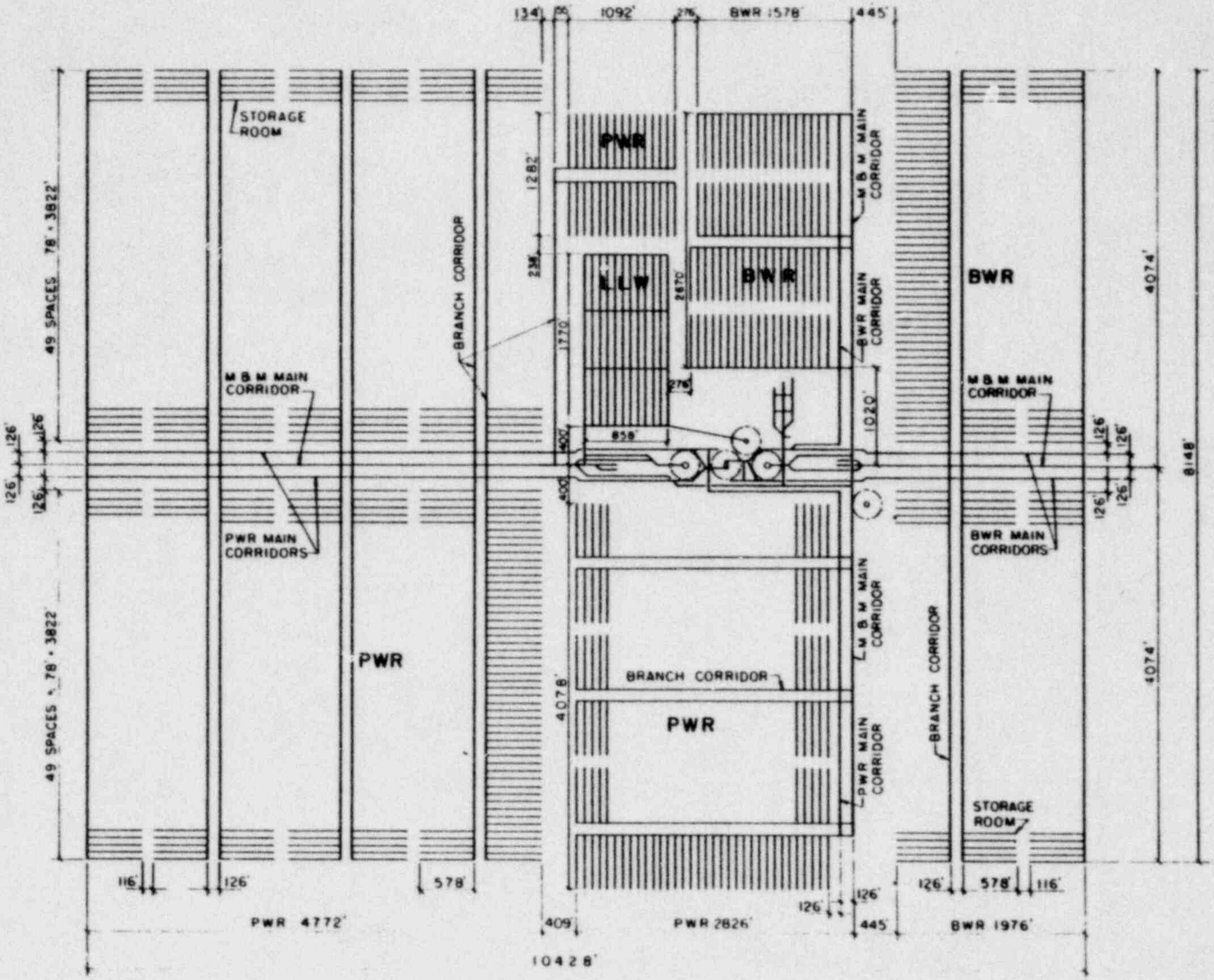

Fig. 2.15. Mine Master Plan for Spent Fuel Repository. (From "Waste Isolation Facility Description for the Spent Fuel Cycle," Office of Waste Isolation, Y/OWI/SUB-77/16543, May 1977.)

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.

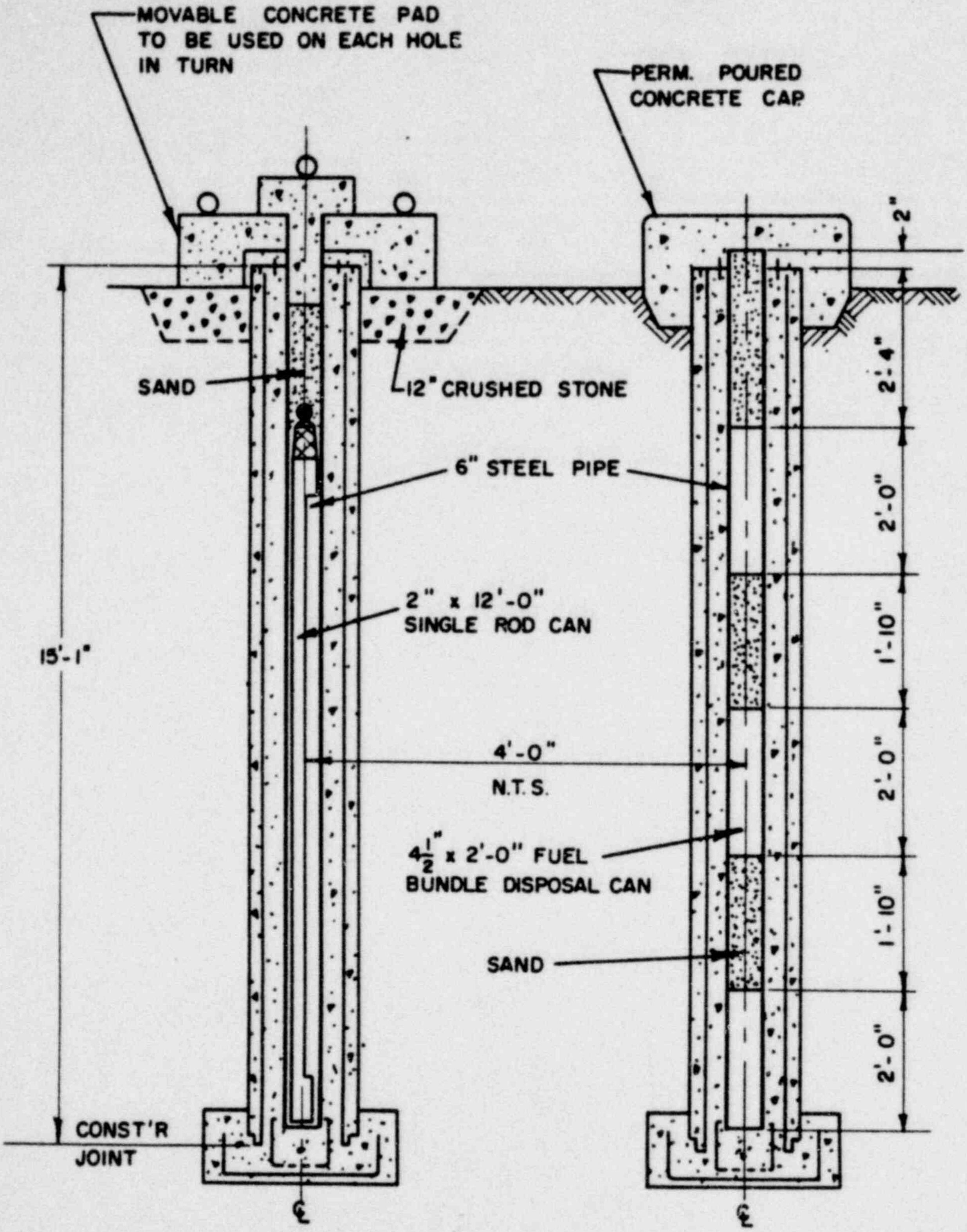

Fig. 2.16. Typical Concrete Tile Hole for the Surface Storage of CANDU Fuel. (Fig. 6 in J. A. Morrison, "AECL Experience in Managing Radioactive Wastes from Canadian Nuclear Reactors," Atomic Energy of Canada Limited, Chalk River Nuclear Laboratories, AECL-4707, March 1974.)

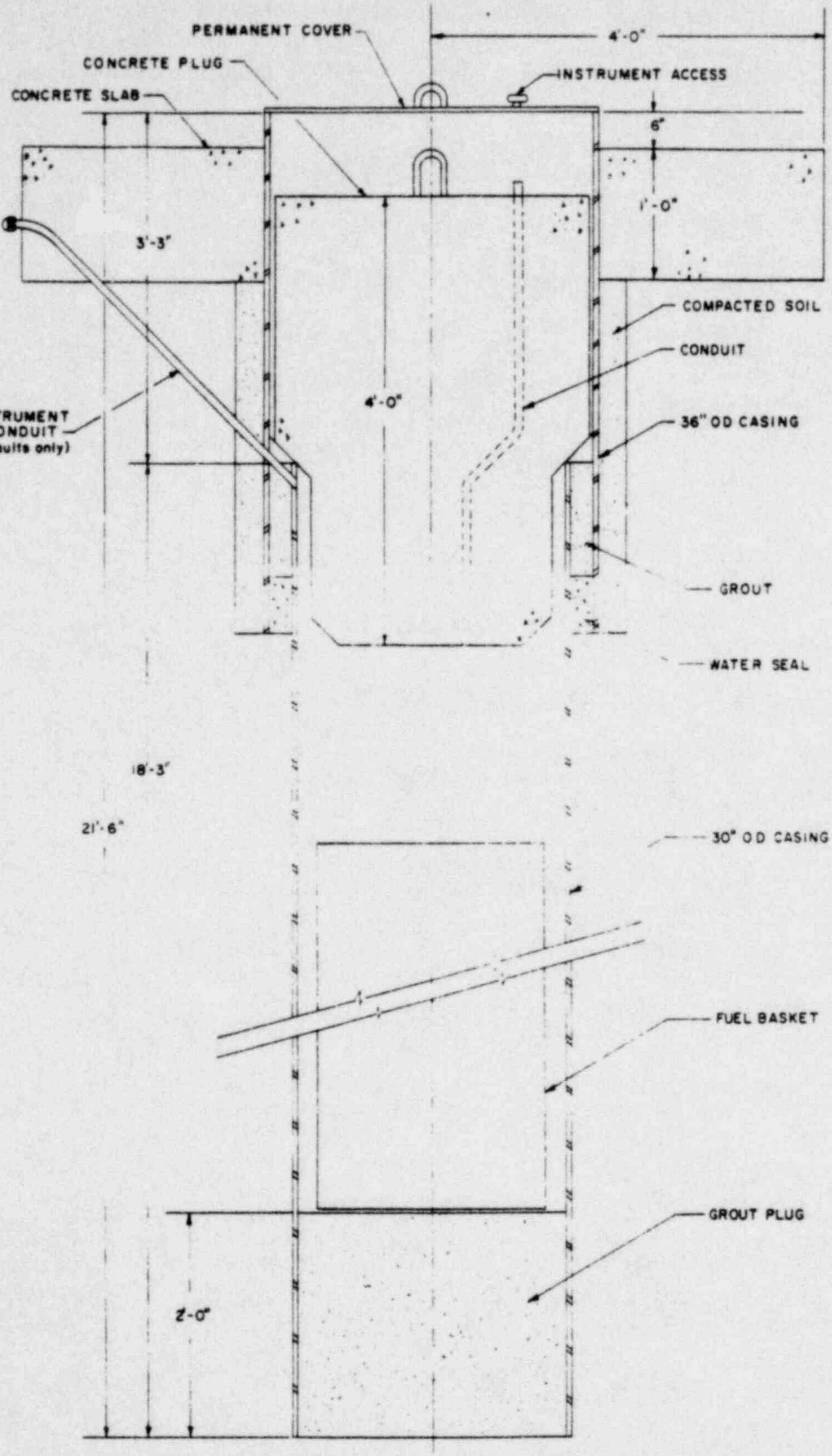

Fig. 2.17. Peach Bottom 1 Spent Fuel Storage Vault. (From J. D. Hammond, R. S. P'Pool and R. D. Morrow, "Safety Analysis Report for Peach Bottom 1 Core 1 Fuel Storage Facility," Idaho Nuclear Corporation for U.S. Atomic Energy Commission, June 1971.)

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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.

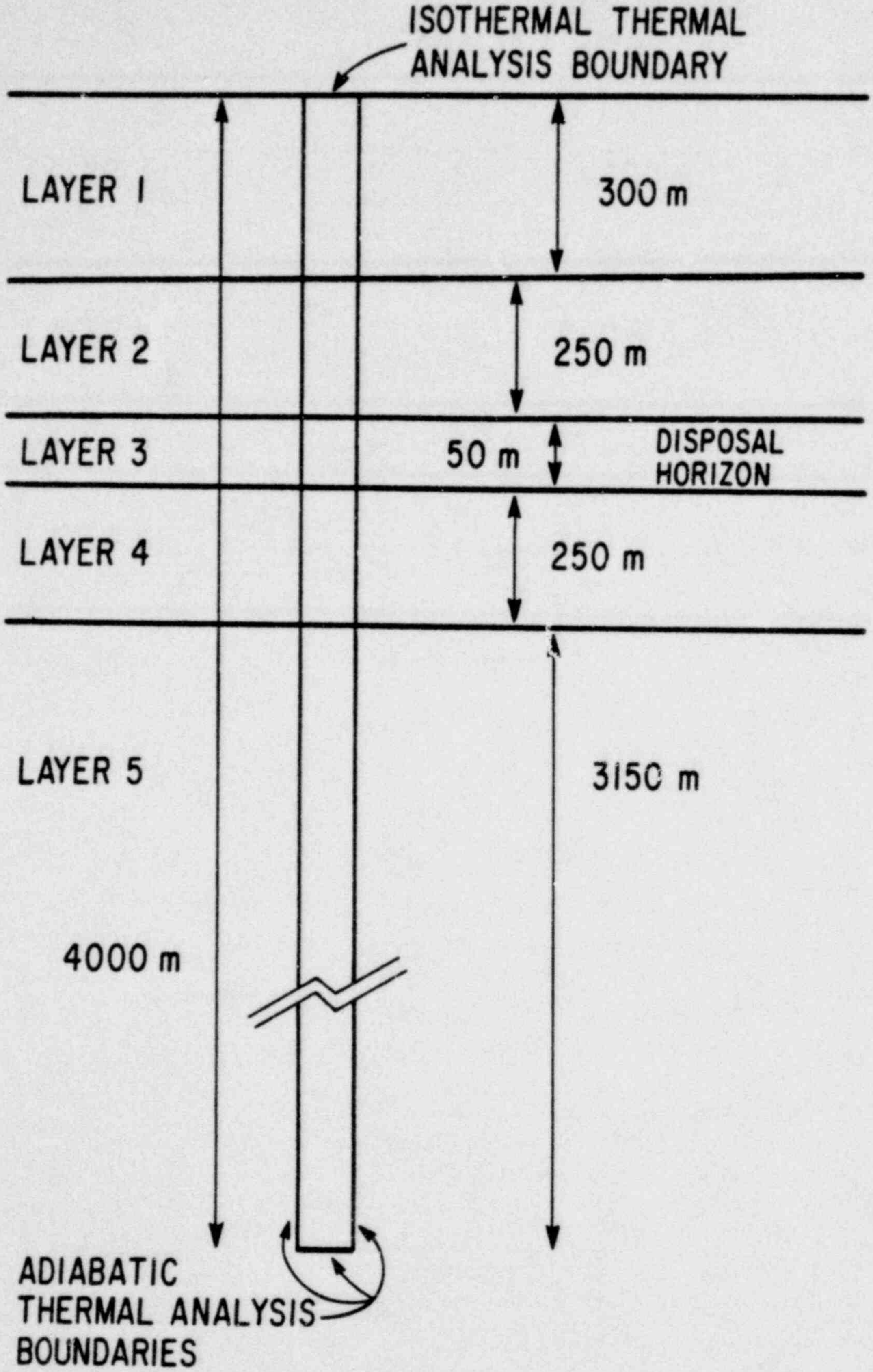

Fig. 3.1. Reference Site and Thermal Model Characteristics.

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 3.34×10^{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.

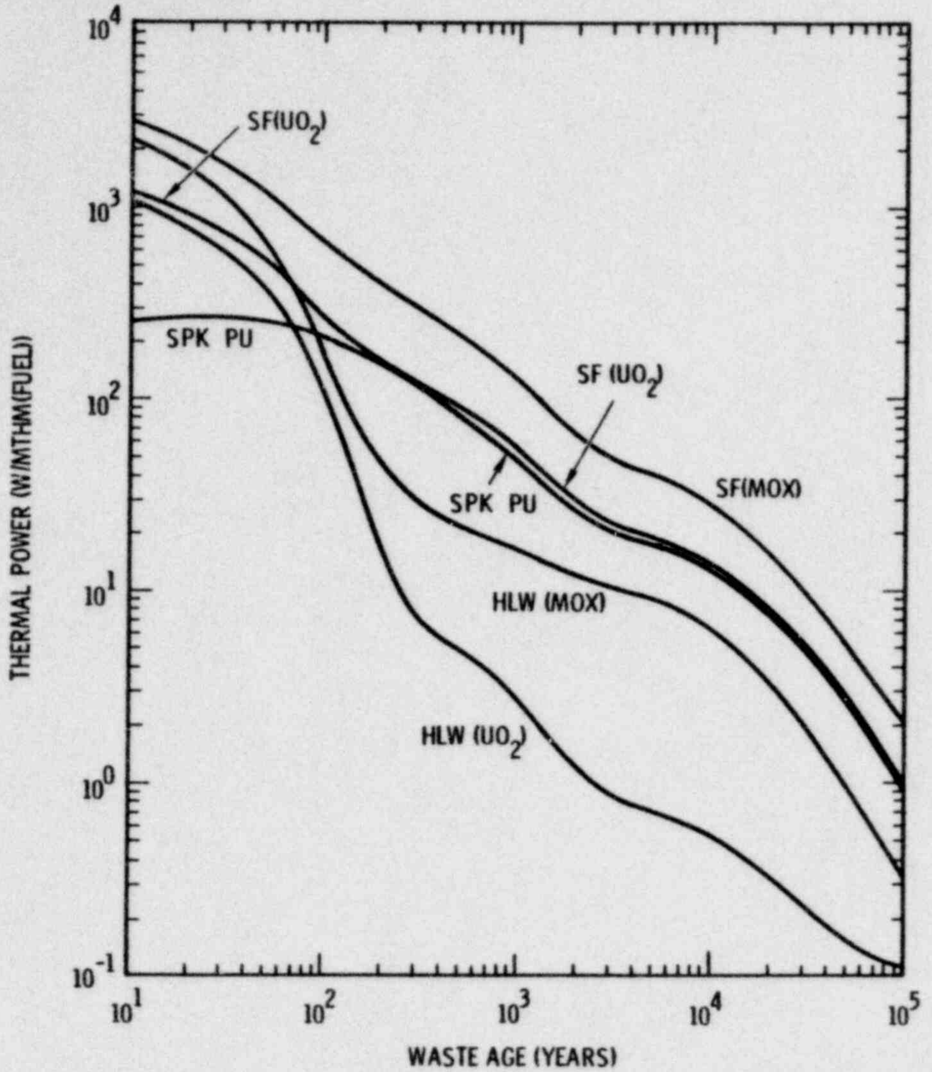

Fig. 3.2. Thermal Power of the Five Waste Types as a Function of Waste Age.

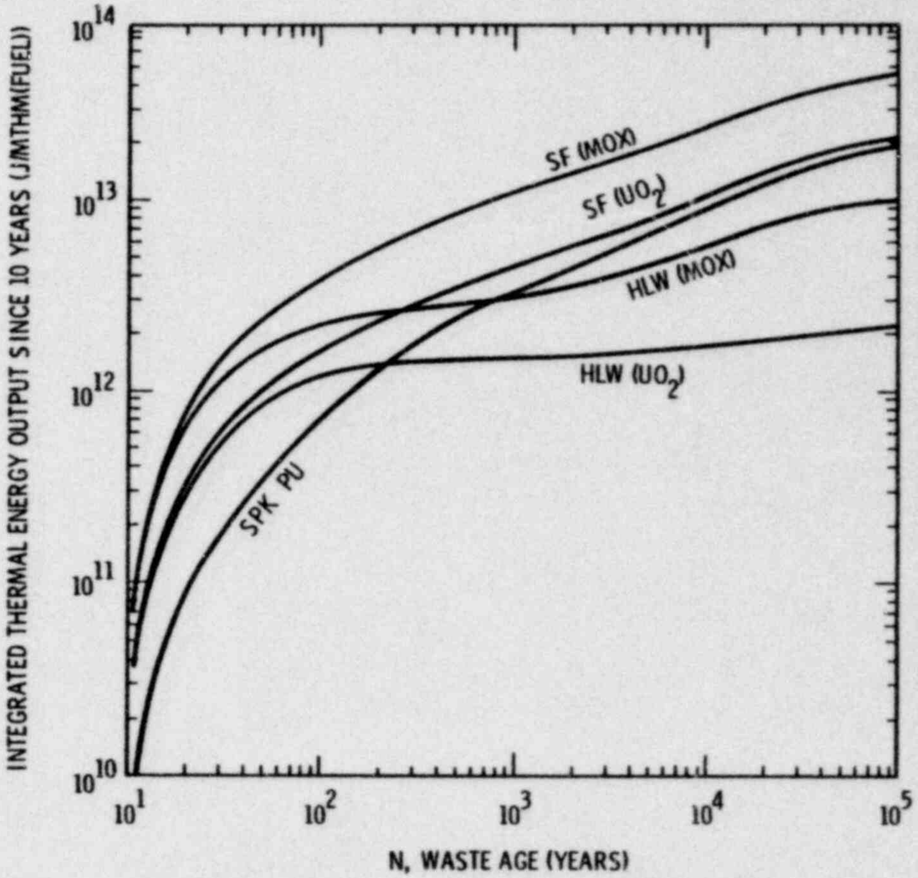

Fig. 3.3. Integrated (10 years to "N" years) Thermal Energy Output of the Five Waste Types.

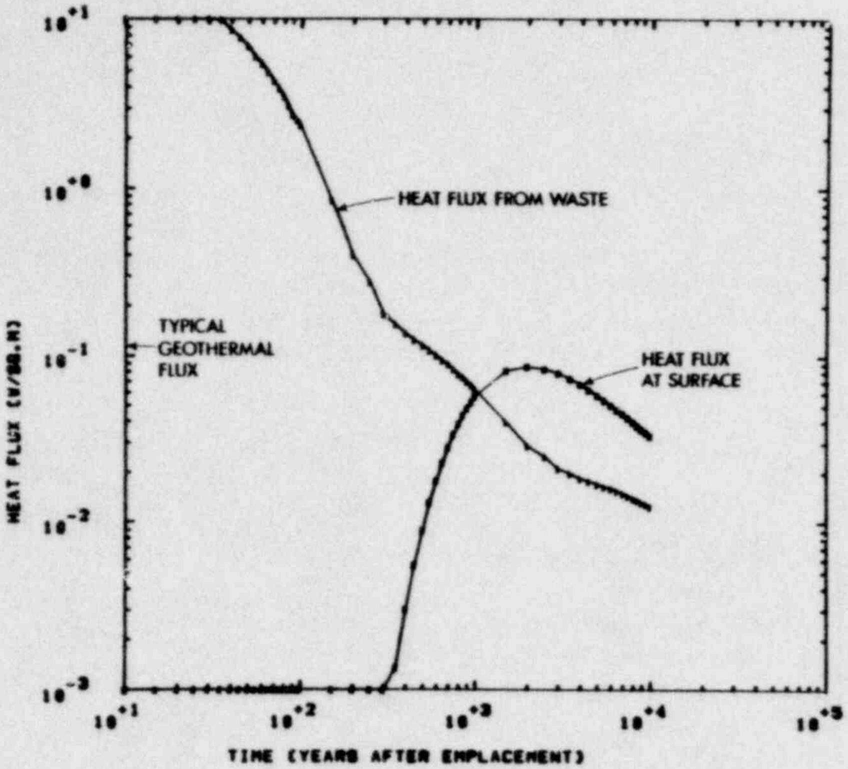

Fig. 3.4. Heat Flux from Waste and Heat Flux at Surface for HLW (UO_2) Emplaced at 106 kW/Acre.

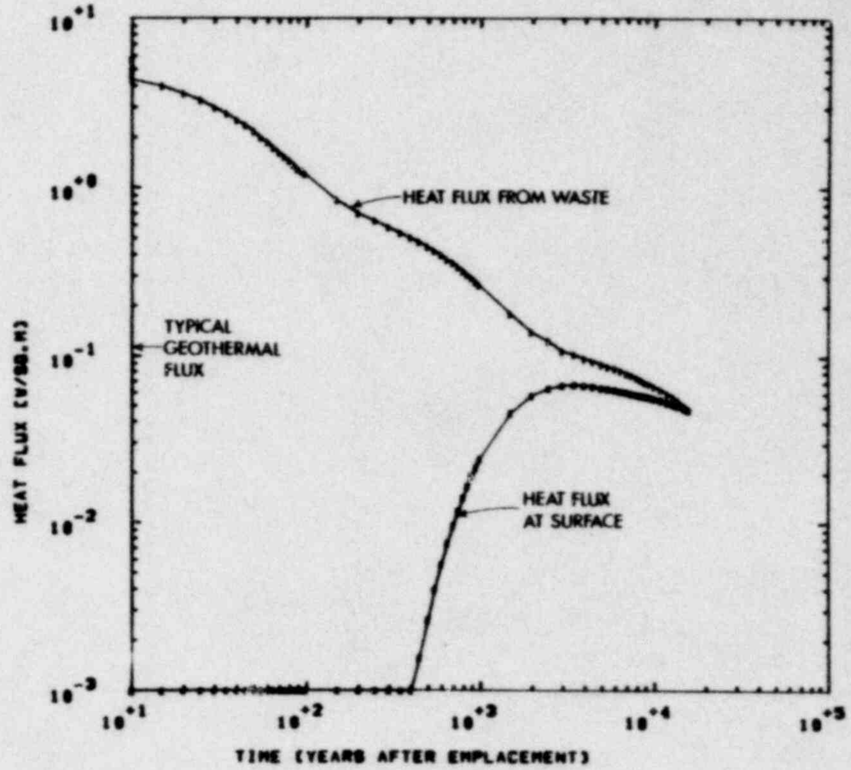

Fig. 3.5. Heat Flux from Waste and Heat Flux at Surface for SF (UO_2) Emplaced at 23.5 kW/Acre.

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.

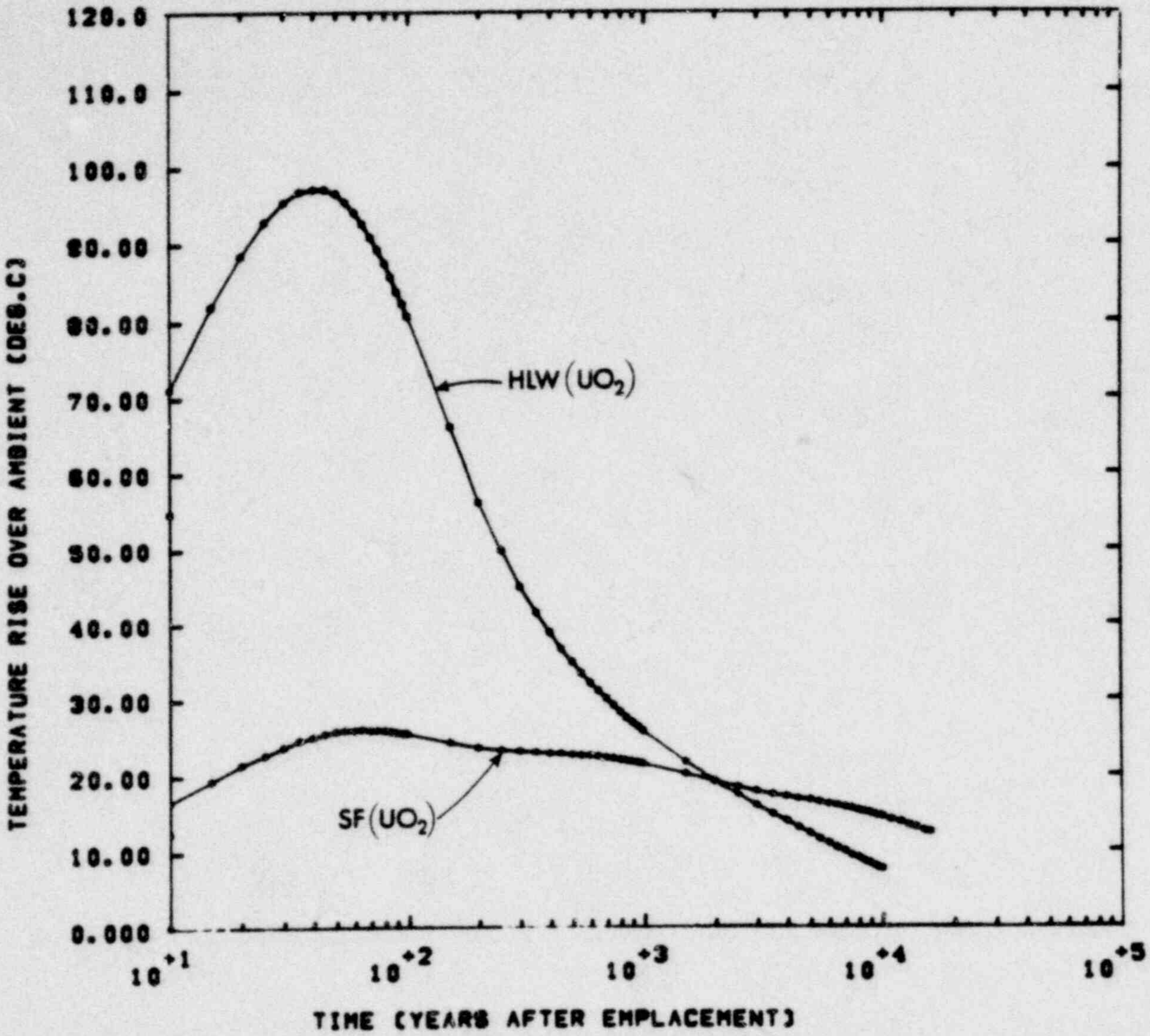

Fig. 3.6. Average Temperature Rise at Waste Disposal Level for HLW (UO_2) Emplaced at 106 kW/Acre and SF (UO_2) Emplaced at 23.5 kW/Acre.

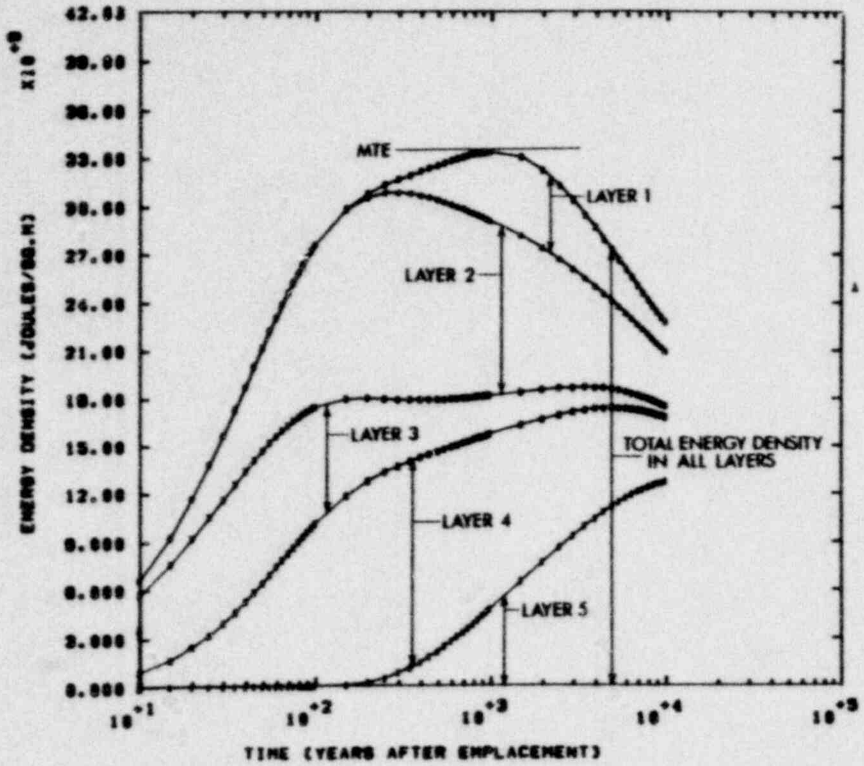

Fig. 3.7. Energy Stored in Geologic Layers for HLW (UO_2) Emplaced at 106 kW/Acre.

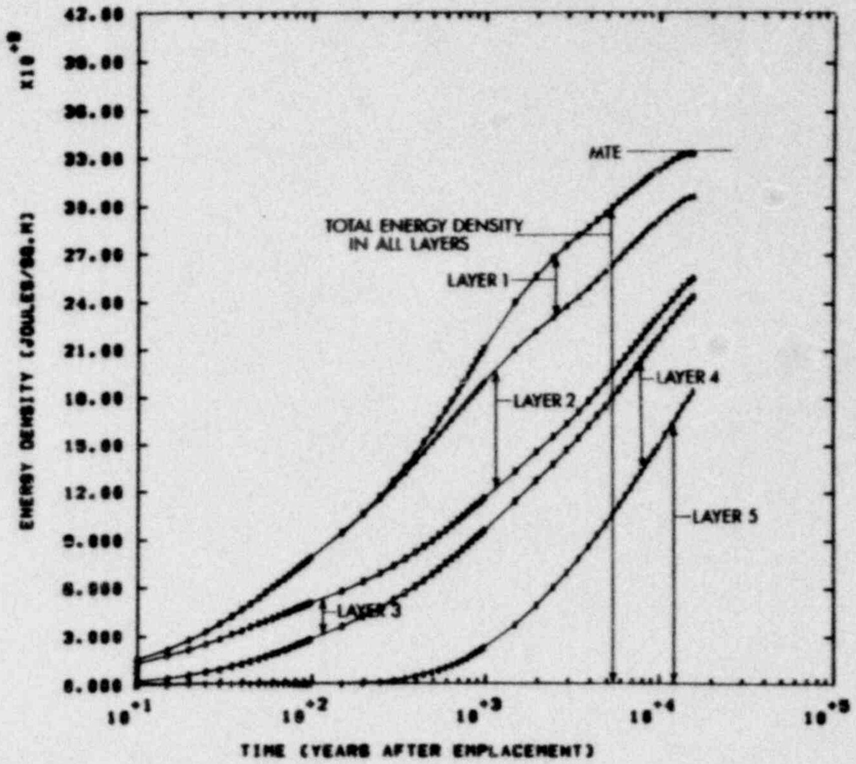

Fig. 3.8. Energy Stored in Geologic Layers for SF (UO_2) Emplaced at 23.5 kW/Acre.

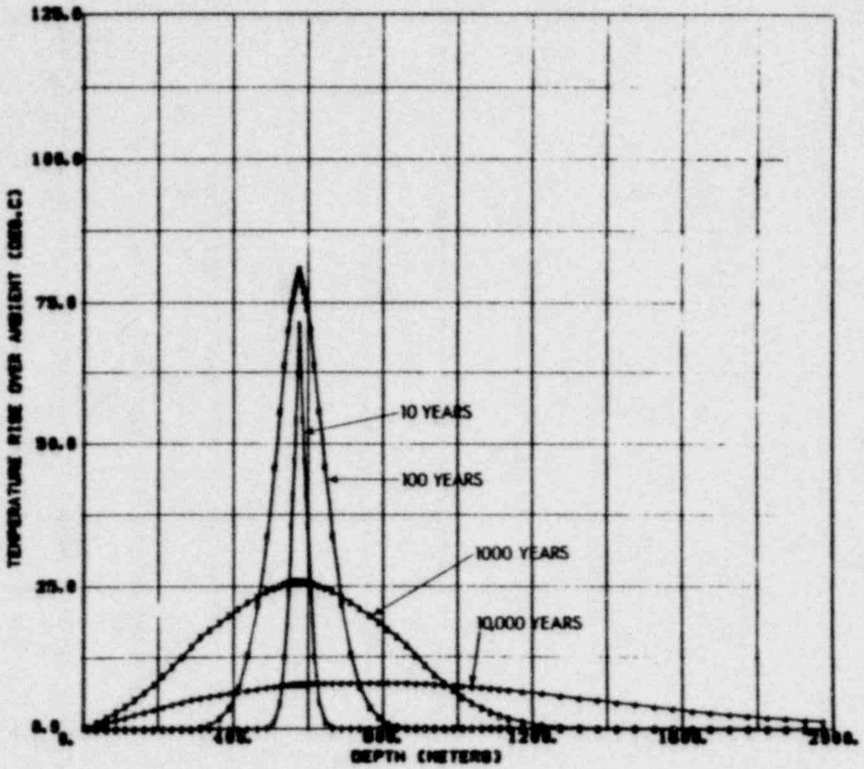

Fig. 3.9. Vertical Temperature Profiles for HLW (UO_2) Emplaced at 106 kW/Acre.

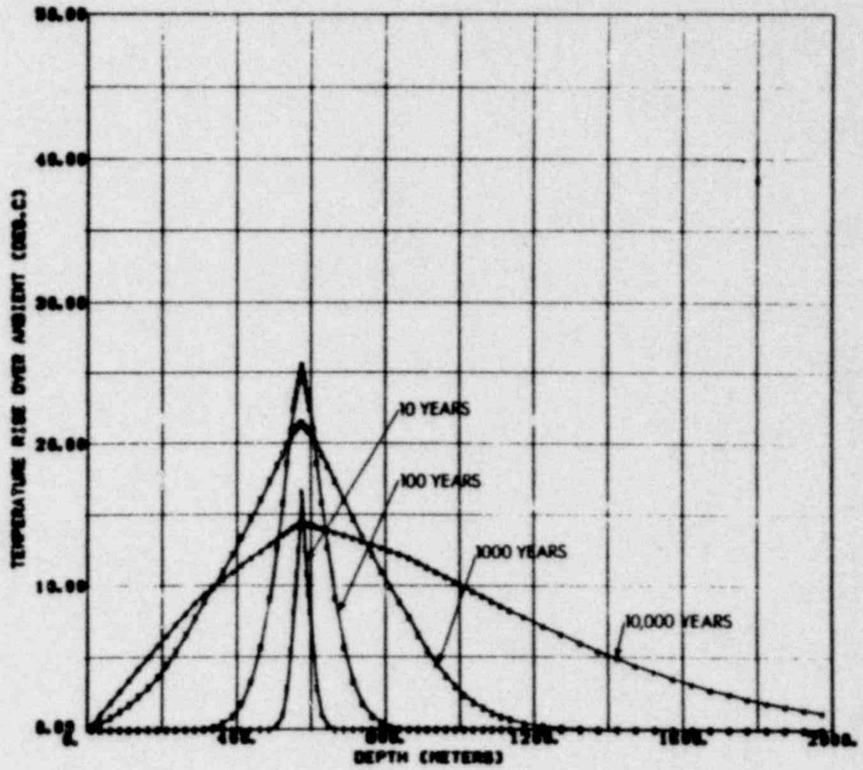

Fig. 3.10. Vertical Temperature Profiles for SF (UO_2) Emplaced at 23.5 kW/Acre.

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	3.3×10^4 MWd/MTHM
Specific power	30 MW/MTHM
Reactor mix, PWR:BWR	2:1
Leakage coefficient, each barrier	1×10^{-3}/yr
Leakage coefficient, each fuel rod	1×10^{-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	1×10^{-3}
Semivolatile transmission	1×10^{-7}
Particulate transmission	1×10^{-9}
Stack height	1×10^2 m
Stack flow	1×10^2 m^3/sec
Stack velocity	8 m/sec
Stack exhaust temperature, above ambient	5 deg K
Center to controlling site boundary	2×10^3 m
Initial specific power	9×10^3 kW/m^2
Surge Pool Basis:	
Activity, total	1×10^{-3} Ci/m^3
Activity, composition	60% Cs-137/134, 25% Co-60/58, 9% H-3, 5% Ni-63, 1% all other β-γ, 0.01% alpha
In-flow	3×10^2 liter/day
Filter flow	4×10^3 liter/min
Cation/anion exchanger flow	2×10^3 liter/min
Heat load	1 MW
Mean fuel load	8×10^2 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	1×10^3/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 1.7×10^7 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	1,250	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.

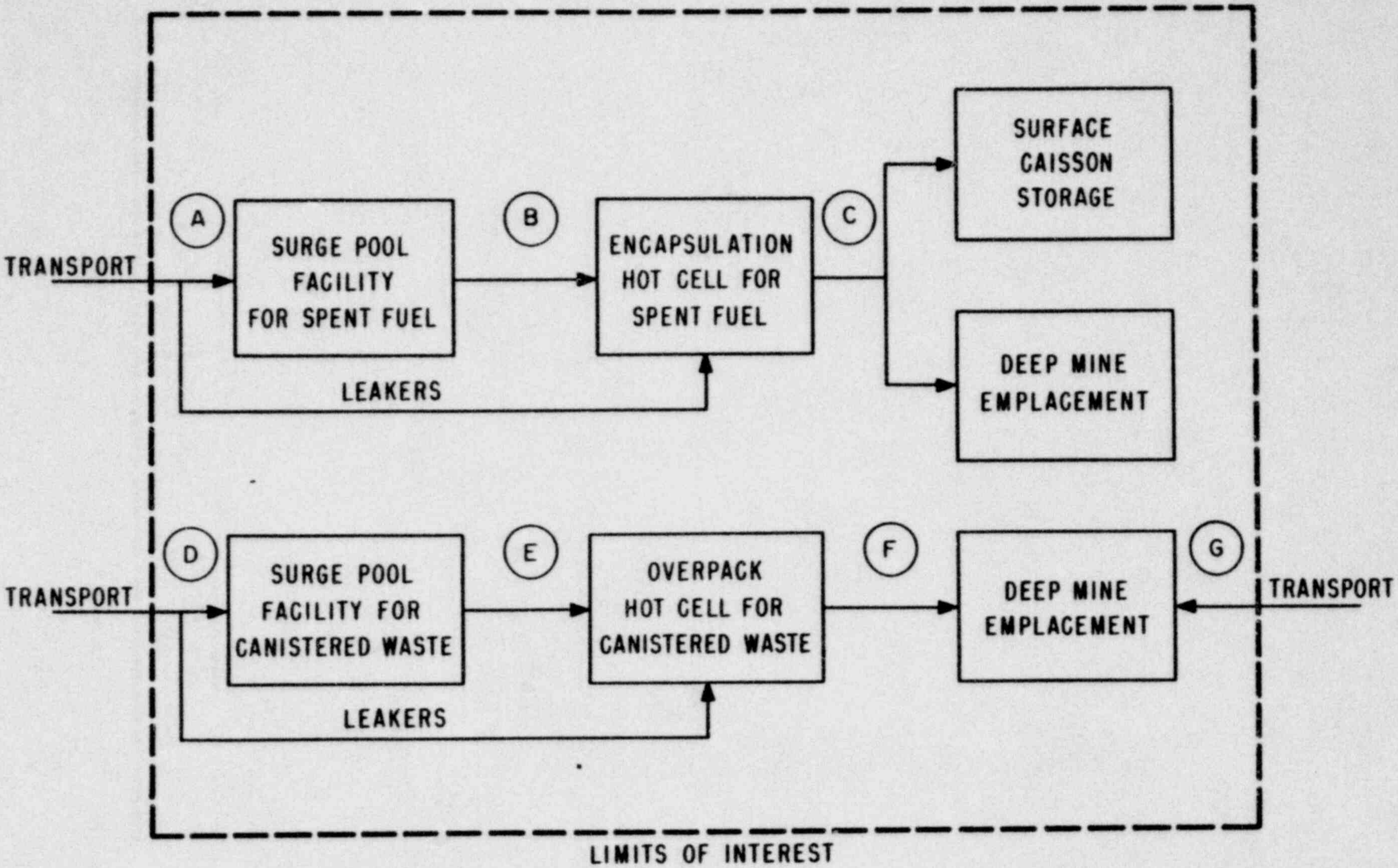

Fig. 4.1. Flow Diagram for Waste Streams in Waste Management Radiological Impact Analysis. (See Table 4.2 for definition of waste streams.)

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	4×10^{-3}	3×10^{-1}[b]	4×10^{3}	40[b]
U-recycle, Pu disposed	3×10^{-3}	1×10^{-1}	3×10^{3}	30
Full recycle	5×10^{-3}	1×10^{-2}	8×10^{2}	6
No-recycle, surface storage	30	7[c]	4×10^{3}	40[c]
No-recycle, deep stowaway	4	5×10^{-1}[b]	3×10^{3}	10[b]
No-recycle, deep throwaway	1	4×10^{-1}	1×10^{3}	10
Background, 9 repositories	6×10^{5}	6×10^{5}	2×10^{2}	6
Background, 14 respositories	1×10^{6}	1×10^{6}	3×10^{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) UO$_2$	Spent Fuel (Ci/MTHM) MOX	High-Level Waste (Ci/canister) UO$_2$	High-Level Waste (Ci/canister) MOX	Spiked PuO$_2$ (Ci/container)
Gases					
H-3	4.04×10^2	4.04×10^2	--	--	--
Kr-85	5.95×10^3	5.95×10^3	--	--	--
I-129	3.71×10^{-2}	3.71×10^{-2}	--	--	--
Total	6.35×10^3	6.35×10^3			
Volatiles					
Cs-134	7.82×10^3	7.82×10^3	2.45×10^4	2.45×10^4	6.84×10^1
Cs-137	8.57×10^4	8.57×10^4	2.68×10^5	2.68×10^5	7.50×10^2
Total	9.35×10^4	9.35×10^4	2.92×10^5	2.92×10^5	8.18×10^2
Particulates					
Sr-90	5.95×10^4	5.95×10^4	1.86×10^5	1.86×10^5	5.20×10^2
Y-90	5.96×10^4	5.96×10^4	1.86×10^5	1.86×10^5	5.21×10^2
Pu-238	5.27×10^2	1.02×10^4	2.71×10^2	2.81×10^3	3.05×10^3
Pu-239	3.24×10^2	4.69×10^2	5.10	2.20×10^1	1.90×10^2
Pu-240	4.83×10^2	1.05×10^3	1.28×10^1	1.47×10^2	2.84×10^2
Pu-241	7.71×10^4	1.84×10^5	1.20×10^3	2.89×10^3	3.87×10^4
Pu-242	1.74	7.91	--	--	8.16×10^{-1}
Am-241	1.80×10^3	4.24×10^3	7.57×10^2	1.65×10^3	7.86×10^2
Am-243	1.73×10^1	2.14×10^2	5.41×10^1	6.70×10^2	--
Cm-242	5.70	7.27×10^1	1.78×10^1	2.28×10^2	--
Cm-244	1.34×10^3	3.37×10^4	4.19×10^3	1.05×10^5	1.48×10^1
Total	2.01×10^5	3.53×10^5	3.78×10^5	4.85×10^5	4.41×10^4

[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 3×10^{-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 6.8×10^{-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 3.7×10^{-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.

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*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	2×10^{-3}	10^{-9}/0.01	2×10^{-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	3.63×10^{-6}	1.15×10^{-4}	4.38×10^{-7}	--	--	--
Filtration system fails	3.63×10^{-6}	1.15×10^{-4}	2.19×10^{-5}	--	--	--
Drop of Heavy Object in SFA Surge Pool	1.82×10^{-5}	5.73×10^{-4}	2.19×10^{-6}	--	--	--
Loss of SFA Surge Pool Cooling						
Filtration system works	5.9×10^{-2}	1.87	7.12×10^{-3}	1.03×10^{-7}	1.62×10^{-6}	1.12×10^{-6}
Filtration system fails	5.9×10^{-2}	1.87	3.56×10^{-1}	1.03	1.62	1.12
Earthquake Damage to SFA Surge Pool	4.21×10^{-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	4.21×10^{-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	3.64×10^{-6}	1.15×10^{-4}	4.38×10^{-7}	6.32×10^{-11}	9.98×10^{-10}	6.90×10^{-10}
Filtration system fails	3.64×10^{-6}	1.15×10^{-4}	2.19×10^{-5}	6.32×10^{-4}	9.98×10^{-4}	6.90×10^{-4}
Drop of SFA in Surface Caisson Facility	2.60×10^{-5}	8.20×10^{-4}	1.56×10^{-4}	--	--	--
Aircraft Impact on SFA Caisson Storage Facility	1.30×10^{-4}	4.09×10^{-3}	1.56×10^{-3}	2.26×10^{-1}	3.36×10^{-1}	2.46×10^{-1}
Drop of SFA in Underground Storage Facility						
Filtration system works	3.64×10^{-6}	1.15×10^{-4}	4.38×10^{-7}	--	--	--
Filtration system fails	3.64×10^{-6}	1.15×10^{-4}	2.19×10^{-5}	--	--	--
Loss of HLSW[a] Pool Cooling						
Filtration system works	--	--	--	1.03×10^{-7}	1.61×10^{-2}	2.86×10^{-7}
Filtration system fails	--	--	--	1.03	1.61	2.86×10^{-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	--	--	--	2.15×10^{-10}	3.37×10^{-9}	5.96×10^{-10}
Filtration system fails	--	--	--	2.15×10^{-3}	3.37×10^{-3}	5.96×10^{-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.

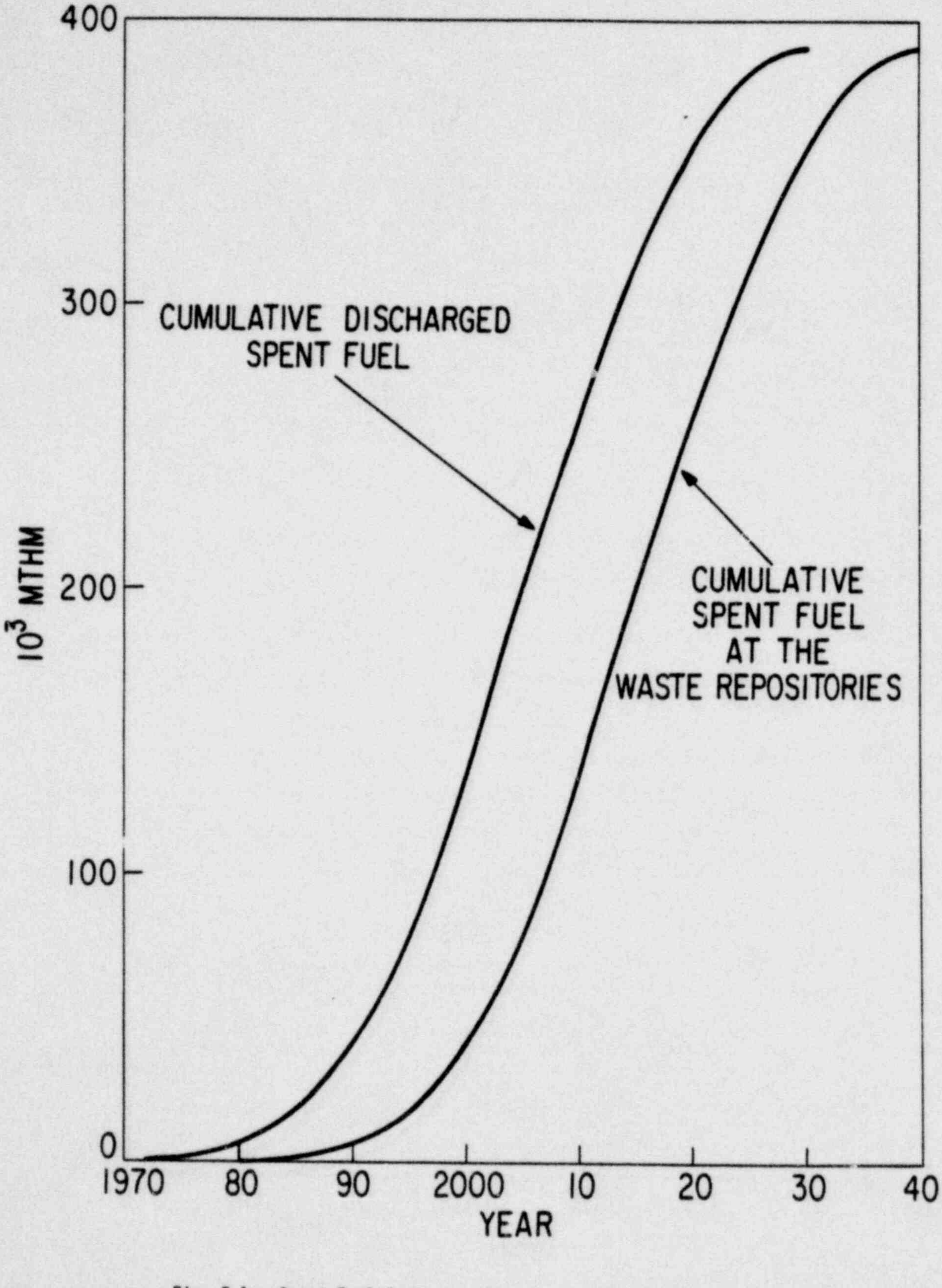

Fig. 5.1. Spent Fuel Inventory for the No-Recycle Options.

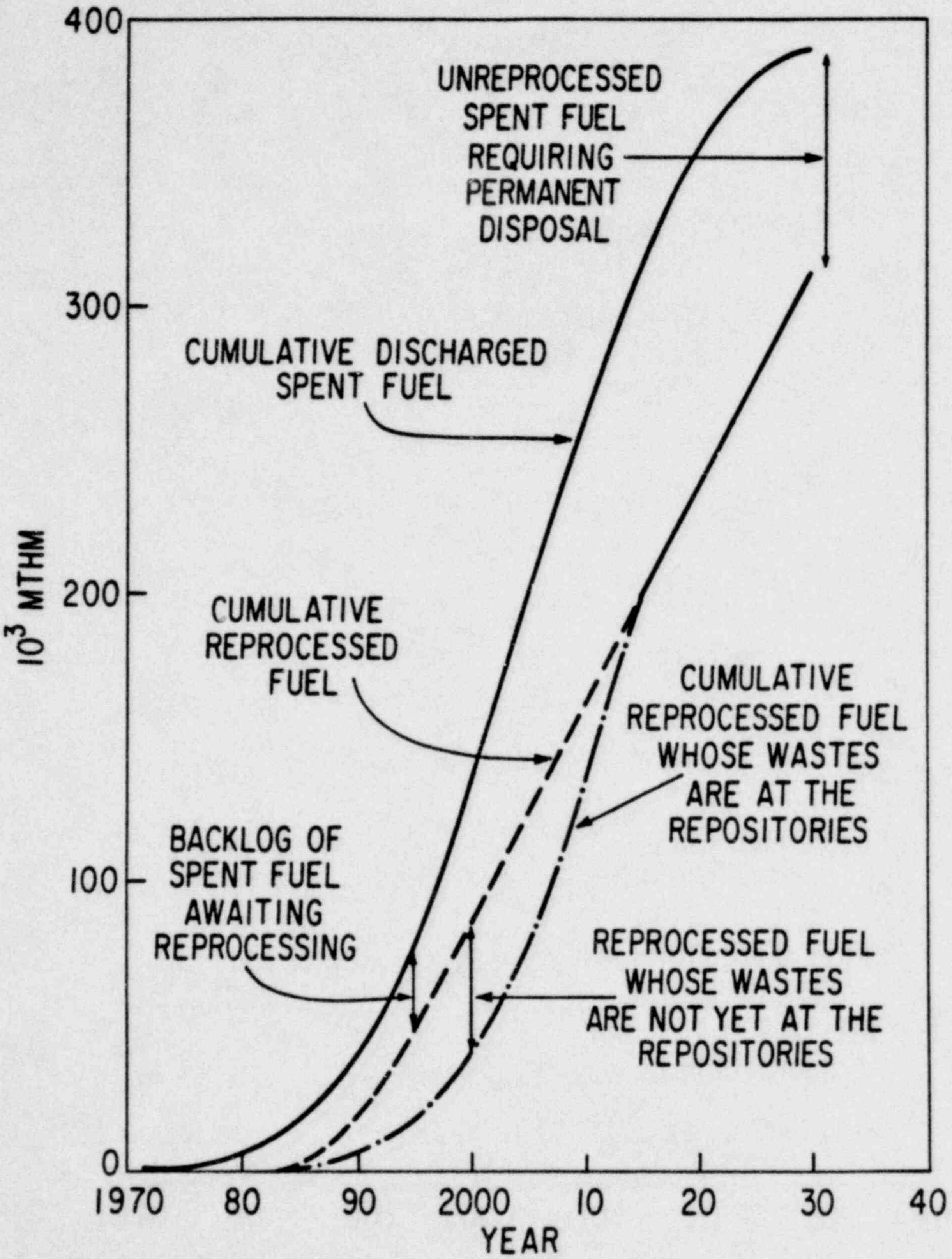

Fig. 5.2. Spent Fuel and Reprocessing Schedule for the U-Only Recycle Options.

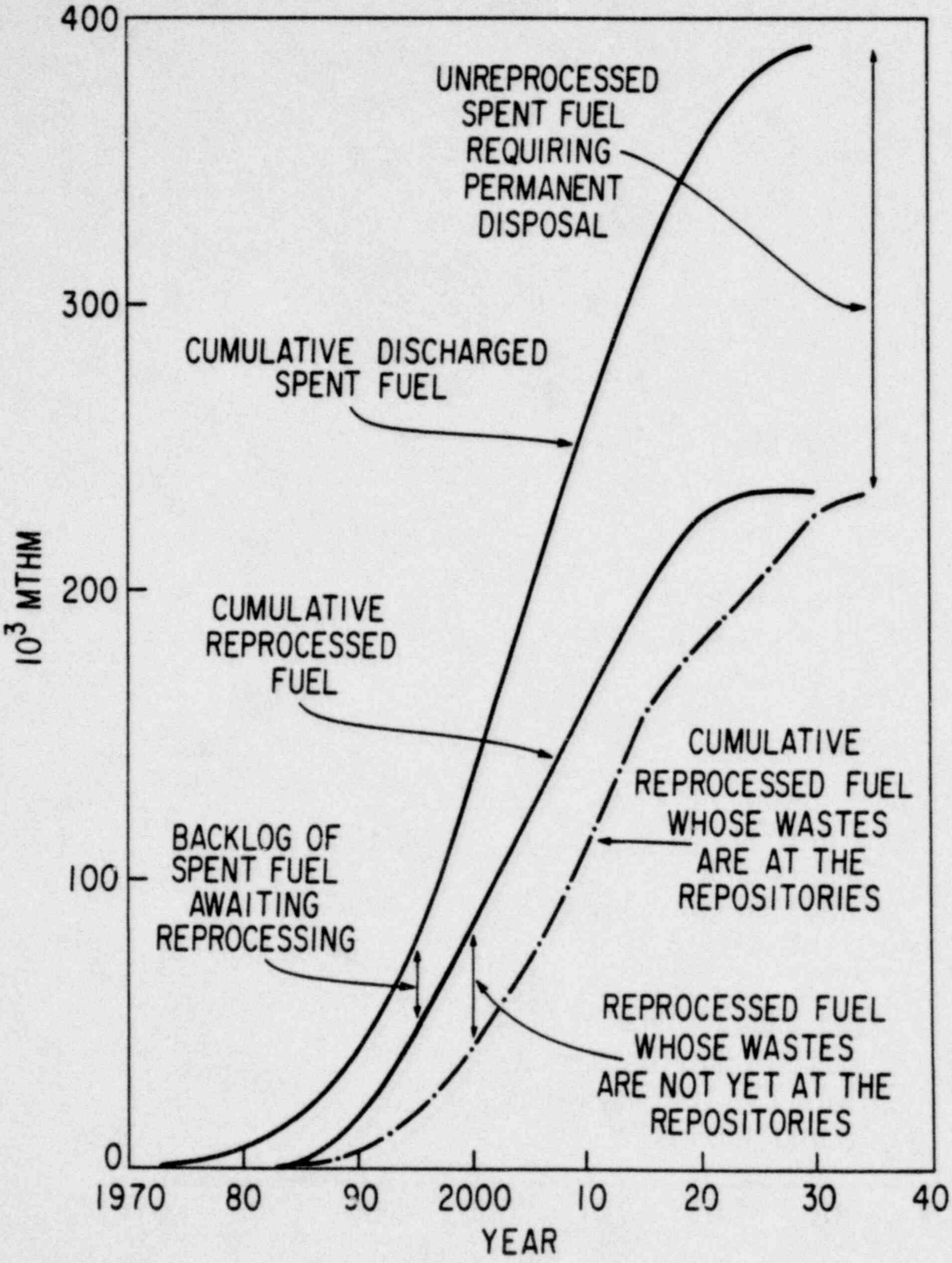

Fig. 5.3. Spent Fuel and Reprocessing Schedule for the Full Recycle Option.

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 Ge..eric 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," Cak 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 above-ground 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 1.3×10^{10} MT (1.4×10^{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 1.3×10^{11} MT (1.4×10^{11} ST). Since the available salt reserves in the United States are estimated at 5.5×10^{13} MT (6.1×10^{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 1.8×10^{11} MT (2.0×10^{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 1.7×10^{11} MT (1.9×10^{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 Reposi- tories	Land Commit- ment (acres)		Fractional Area[a] of Hanford and Savannah River Sites[b]		Salt Commit- ment (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	1.3×10^{11}	0.23%
U-only recycle	14	28,000	70,000	0.08	0.15	1.8×10^{11}	0.33%
Full recycle	13	26,000	65,000	0.07	0.14	1.7×10^{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: Assemblies	MT	Deep, Geologic Repositing of Spent Fuel[a]: 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

	Option					
Waste Type	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		1,300			
Hulls	40		320			
TRU-ILW	320	160	2,300	1200	470	3,600
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		400			
HLSW - MOX	20		1,500			
Hulls	40	234	200	1610	580	6,050
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]	3.5×10^{13}	8.7×10^{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 = 1×10^6 kWe-year = 1000 MWe-year.

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

	Type of Repository[b]								
	Spent Fuel[a]			HLSW + Hulls + ILW + LLW			Plutonium[c]		
Period	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 28-subsequent	4
Und rground 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 [$(\$7.15 \times 10^9) \div (3.9 \times 10^8\ kg)$]. It would cost an additional \$8/kg [$(\$3.0 \times 10^9) \div (3.9 \times 10^8 kg)$] 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	3.88E-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	4.75E-03	1.35E+00	4.71E-07
NB 95	1.71E-16	[illegible].72E-12	3.24E-17
NB 95M	1.79E-19	6.38E-14	9.16E-20
TC 99	8.37E+02	1.[illegible]E+01	2.43E-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	4.93E-07	2.19E-03	2.31E-09
SN121M	4.15E-01	1.62E+01	1.72E-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-06	5.85E-01	7.61E-05
SB126M	7.52E-09	5.91E-01	4.00E-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	5.01E+00	7.32E+03	8.20E-02
CS135	4.27E+02	3.77E-01	1.33E-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	[illegible].03E-05
PR144	1.30E-06	9.82E+01	7.61E-04
PM147	6.94E+00	6.44E+03	3.32E-03
SM151	3.30E+01	1.03E+03	1.30E-03
EU152	2.73E-02	5.46E+00	9.73E-05
EU154	5.49E+01	7.90E+03	6.55E-02
EU155	4.34E-02	5.54E+01	4.65E-05
GD153	1.73E-07	6.00E-04	8.77E-10
TB160	1.50E-17	1.69E-13	1.44E-18
HO165M	5.59E-04	1.00E-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	6.42E-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	3.08E-17	1.01E-09	2.46E-15
BI210	7.96E-16	9.90E-11	2.61E-16
BI211	3.99E-22	1.61E-13	6.28E-19
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	8.44E-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	3.15E-08	1.53E-12
PO214	<1.E-20	1.01E-09	4.60E-14
PO215	<1.E-20	1.61E-13	7.06E-18
PO216	4.70E-16	1.64E-04	6.69E-09
PO218	3.57E-19	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-18
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-18	1.61E-13	5.60E-18
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-19	2.41E-13	1.25E-18
TH227	5.09E-18	1.61E-13	5.54E-18
TH228	1.99E-07	1.63E-04	5.35E-09
TH229	1.49E-07	3.15E-08	9.64E-13
TH230	2.55E-05	4.96E-07	1.40E-11
TH231	2.99E-14	1.58E-08	1.25E-14
TH232	5.85E-06	6.39E-13	1.55E-17
TH234	6.72E-16	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	4.28E-06	9.16E-05	2.94E-09
U233	4.79E-03	4.63E-05	1.35E-09
U234	1.06E+00	6.67E-03	1.92E-07
U235	7.39E-03	1.56E-08	4.40E-13
U236	2.05E+01	1.30E-03	3.53E-08
U237	1.13E-07	9.24E-03	6.13E-09
U238	4.01E-05	1.34E-11	3.35E-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	3.84E+02	1.60E-05
PU242	2.23E+00	8.71E-03	2.57E-07
AM241	7.06E+01	2.42E+02	8.98E-03
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-06
CM246	3.02E-02	9.32E-03	3.06E-07

POOR ORIGINAL

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

Nuclide	Grams MTHM(fuel)	Curies MTHM(fuel)	kW MTHM(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	4.75E-03	1.35E+00	4.71E-07
NB 95	1.71E-16	6.72E-12	3.24E-17
NB 95M	1.73E-19	6.54E-14	3.16E-20
TC 99	8.37E+02	1.44E+01	2.43E-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	1.73E-05
SN123	3.40E-10	2.89E-06	9.85E-12
SN126	2.18E+01	5.91E-01	6.38E-07
SB125	6.55E-01	6.95E+02	2.32E-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	1.53E-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	7.34-137
XE131M	1.70E-94	1.43E-89	2.79E-95
XE133	2.10-208	3.92-203	6.13-209
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	8.03E-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	2.73E-02	5.46E+00	3.78E-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	1.08E-08

Table A.2. SF(UO_2) Continued

Nuclide	Grams MTHM(fuel)	Curies MTHM(fuel)	kW MTHM(fuel)
TL207	1.69E-13	3.71E-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-16
PB211	1.30E-18	3.22E-11	1.06E-16
PB212	1.16E-08	1.61E-02	2.32E-08
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-12
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-16	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-16	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	8.10E-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-09	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	8.56E-04	1.83E-02	5.66E-07
U233	4.83E-03	4.63E-05	1.35E-09
U234	1.52E+02	9.42E-01	2.71E-05
U235	1.48E+00	3.16E-06	8.78E-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-03	1.85E+00	1.23E-06
U238	9.42E+05	3.14E-01	7.94E-06
NP237	6.46E+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	5.26E+03	3.24E+02	1.01E-02
PU240	2.19E+03	4.[illegible]3E+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-06
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	8.68E+02	5.11E-03
Y 91	1.70E-20	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	4.96E-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	9.50E-15
TE127M	6.51E-13	6.15E-09	3.33E-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	1.90E-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	8.04E-01	1.17E+02	9.58E-04
EU155	6.34E-04	8.08E-01	6.80E-07
GD153	2.52E-09	8.89E-06	1.29E-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	4.16E-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	6.36E-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	3.34E-19
PO212	<1.E-20	7.83E-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	6.13E-14	9.45E-09	3.08E-13
FR221	1.07E-19	1.90E-10	7.07E-15
FR223	2.64E-20	1.01E-12	2.37E-19
RA223	1.37E-15	7.05E-11	2.45E-15
RA224	7.63E-12	1.22E-06	4.16E-11
RA225	4.88E-15	1.92E-10	1.26E-16
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	3.65E-17
AC228	6.99E-21	1.67E-14	[illegible].31E-20
TH227	2.22E-15	7.02E-11	2.42E-15
TH228	1.48E-09	1.22E-06	3.96E-11
TH229	9.06E-10	1.94E-10	5.36E-15
TH230	3.36E-04	6.53E-06	1.65E-10
TH231	8.27E-12	4.39E-06	3.46E-12
TH232	4.67E-07	5.10E-14	1.23E-19
TH234	1.01E-09	2.35E-05	8.36E-12
PA231	1.27E-08	6.05E-10	1.85E-14
PA233	5.59E-07	1.14E-02	1.54E-09
PA234	1.18E-14	2.35E-08	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

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Table A.3. SPK Pu Continued

Nuclide	Grams MTHM(fuel)	Curies MTHM(fuel)	kW MTHM(fuel)
U237	1.90E-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	2.68E-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-05
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-09

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	1.03E-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	1.79E-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	2.05E+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	6.38E-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	4.00E-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	6.01E+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	7.98E+03	6.55E-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	1.00E-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	6.18E-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	4.88E-14
BI214	2.70E-15	1.21E-07	1.68E-12
PO210	4.45E-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	7.73E-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	8.73E-05	3.41E-09
PO218	4.26E-16	1.21E-07	4.37E-12
AT217	4.86E-21	7.90E-09	3.71E-13
RN219	1.97E-13	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	3.03E-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

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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
U238	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.[illegible]E+02	4.71E+01	1.47E-03
PU241	9.08E+00	9.23E+02	3.63E-05
PU242	1.02E+01	3.99E-02	1.18E-06
AM241	1.53E+02	5.26E+02	1.75E-02
AM242	1.[illegible]9E-04	8.83E+01	1.16E-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 MTHM(fuel)	Curies MTHM(fuel)	kW MTHM(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	4.79E-03	1.35E+00	4.71E-07
NB 95	1.71E-16	6.72E-12	3.24E-17
NB 95M	1.73E-19	5.58E-14	9.16E-20
TC 99	8.37E+02	1.44E+01	2.43E-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	1.70E-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-06	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	1.53E-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-06
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	6.13-209
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	7.03E+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	1.08E-08

Table A.5. SF(MOX) Continued

Nuclide	Grams MTHM(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	1.16E-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
AC229	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 MTHM(fuel)	Curies MTHM(fuel)	kW MTHM(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-[illegible]	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	9.06E+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

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	8.3×10^{-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

Year	Gigawatts			Assemblies Discharged						MT U Discharged	
				Annual			Cumulative				
	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	4[illegible]	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	3.8	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	801
1975	6.1	35.7	18.8	815	724	1539	2151	1909	4060	489	1290
1976	8.3	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	8.0	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	3913	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	41898	38127	81025	4056	25737
1988	21.0	222.0	180.0	7800	6933	14733	49698	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	69040	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	14650	13022	27672	553126	491661	1044787	8790	331874
2018	0.0	288.8	309.8	13740	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	255.8	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	4070	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	2109	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 Reprocessed	MT Uranium Spent Fuel Remaining	Spent Fuel Assemblies Remaining BWR	Spent Fuel Assemblies Remaining PWR	Spent Fuel Assemblies Remaining 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

POOR ORIGINAL

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	92333	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	75902	79381	448480	594140	3356723	236682	1417257
2022	239705	2510	418333	14031	79271	81826	462294	612440	3460113	243972	1460910
2023	246867	2585	430833	14450	81639	84271	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	[illegible]	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	3978	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

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	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	89151	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

POOR ORIGINAL

REACTOR WASTE MANAGEMENT SCHEDULE - Schedule 6

MOX Fuel Status

	Annual MT Reprocessed			Annual MT Produced				Spent MOX3 Assemblies					
								Annual			Cumulative		
Year	UOX	MOX1	MOX2	MOX1	MOX2	MOX3	GW OK for MOX	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	600.0	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	2170.4	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	28507
2008	3410.9	1419.4	2661.7	2392.1	1558.8	2698.5	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	1928.7	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	2852	6060	52790	46925	99715
2017	1484.3	2985.5	1928.7	2653.0	2592.6	1266.9	265.8	2722	2420	5142	55512	49345	104857
2018	530.6	3024.5	2038.5	2191.0	2510.9	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	88136	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 (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	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	41826	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	87	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	259887	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	7989	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	18702	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	390684	709253	4007077	2385896	14286801
2018	5604	31659	6697	37937	70975	400986	731388	4132137	2460359	14732691
2019	5604	31659	6967	39364	72535	409801	751247	4244336	2527165	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

	HLW - UOX		HLW - MOX		HULLS		TRU - ILW		TRU - LLW	
Year	(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	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	1710	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	252197
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	16228	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	382676
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

	HLW - UOX		HLW - MOX		HULLS		TRU - ILW		TRU - LLW	
Year	(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	28	158	0	0	161	910	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	8216	0	0	8390	47403	63514	358835	213657	1279386
1998	1719	9710	14	82	9999	56494	76034	429569	255774	1531583
1999	2014	11378	43	245	11870	67061	90872	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	501271	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	817883	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	22870	55465	313364	535769	3026944	1802304	10792242
2018	5604	31659	4270	24123	56969	321858	554664	3133699	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 (0.4×10^{-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

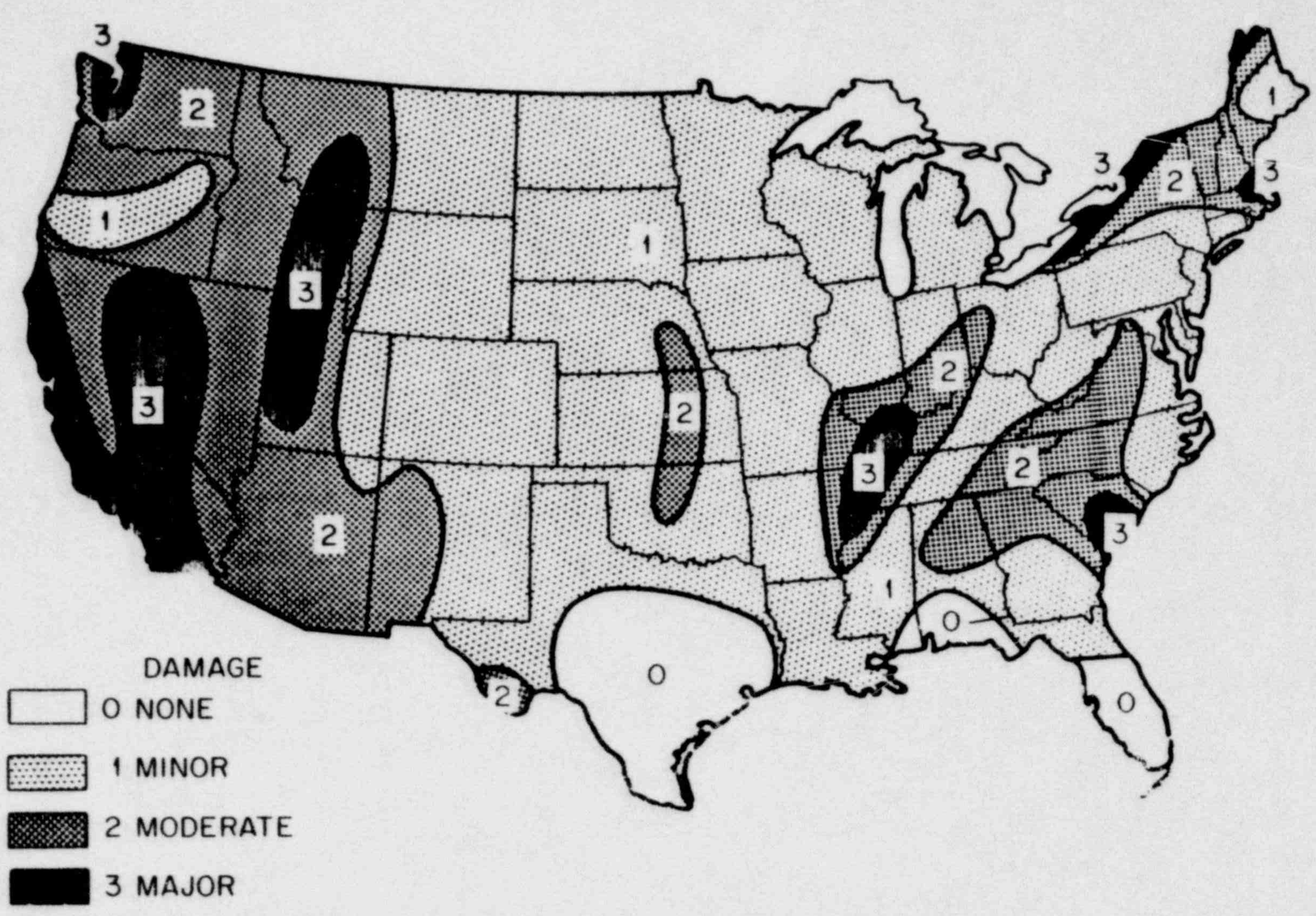

Fig. D.1. Seismic Risk Map of the United States. (Fig. II-4 in "Final Environmental Statement, Management of Intermediate-Level Radioactive Waste," U.S. Energy Research and Development Administration, ERDA-1553, September 1977.)

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 ρ is the bulk density and ε 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, $K = 1 + K_aR_f$. 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.

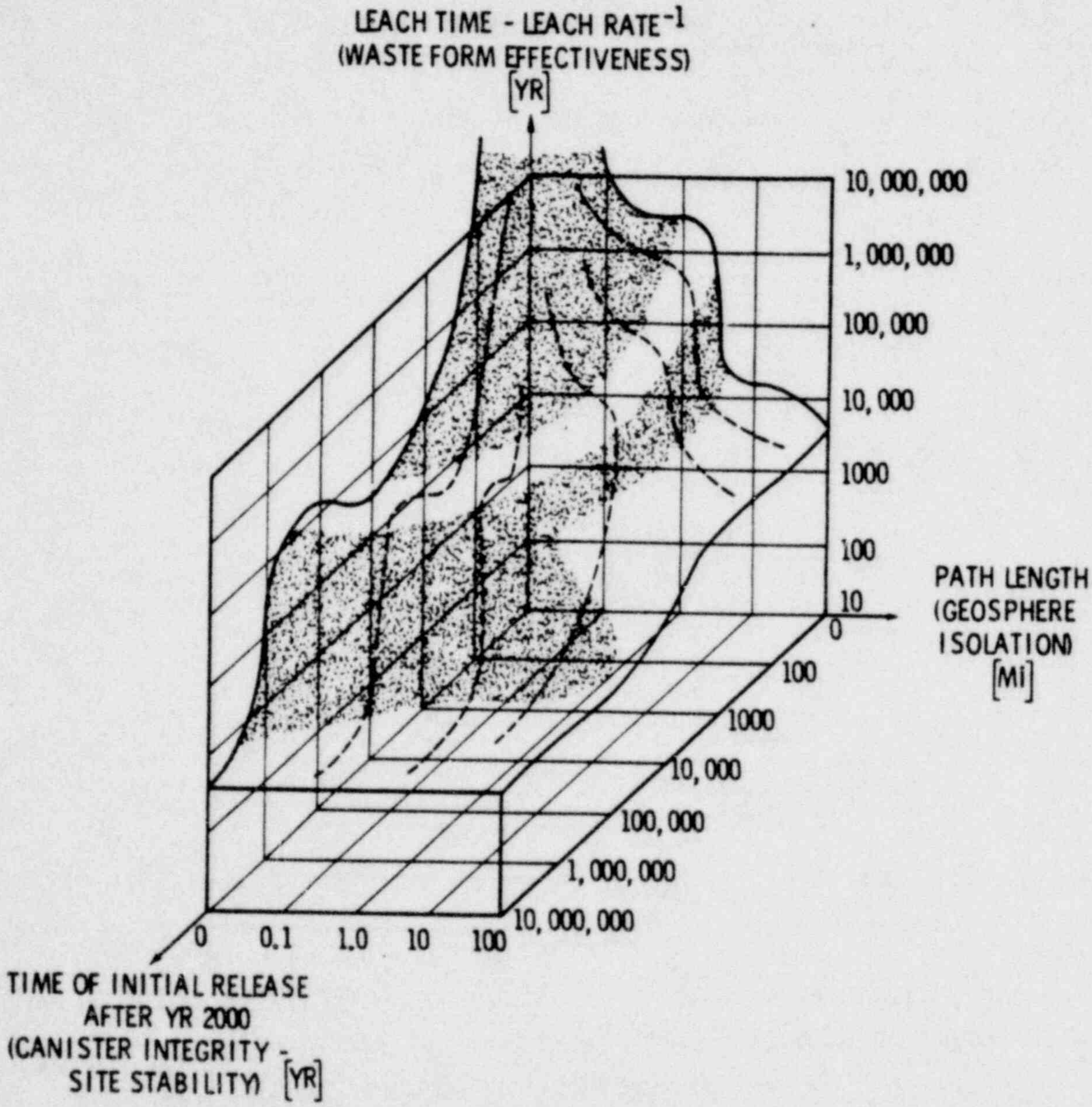

Fig. D.2. Waste Management Control Surface for Incremental Background Dose with No Partitioning. (Adapted from Alternatives for Managing Wastes from Reactors and Post-Fission Operations in the LWR Fuel Cycle," U.S. Energy Research & Development Administration, ERDA 76-43, May 1976.

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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