RADIOACTIVE WASTE MANAGEMENT

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The management of radioactive waste is a very important part of the nuclear industry. The future of the nuclear power industry depends to a large extent on the successful solution of the perceived or real problems associated with the disposal of both low-level waste (LLW) and high-level waste (HLW). All the activities surrounding the management of radioactive waste are reviewed. The federal government and the individual states are working toward the implementation of the Nuclear Waste Policy Act and the Low-Level Waste Policy Act. The two congressional acts are reviewed and progress made as of early 1990 is presented. Spent-fuel storage and transportation are discussed in detail as are the concepts of repositories for HLW. The status of state compacts for LLW is also discussed. Finally, activities related to the decommissioning of nuclear facilities are also described.

I. INTRODUCTION

This paper is not intended as a critical review but as an overview of the state of radioactive waste management in early 1990. The review is based on the recent congressional acts, the most recent government reports, numerous other technical references, and the authors' general knowledge and experience of many years in the nuclear field.

With respect to waste, the nuclear industry is no exception to other industries as waste is produced that must be properly controlled to protect people and their environment. Radioactive waste, however, differs from other industrial wastes in two respects. First, the risk that it poses decreases with time, thanks to decay. Second, the volume of nuclear waste is much smaller than that of other industrial wastes. As an example, compare the emissions of a coal-fired plant to those of a nuclear power plant. A 1000-MW(electric) coal plant burns ~11 000 t of coal every 24 h, and it discharges directly into the atmosphere ~300 t of SO_2 and ~5 t of fly ash containing small quantities of such elements as chlorine, cadmium, arsenic, mercury, lead, and many radioactive elements. A 1000-MW(electric) nuclear plant produces ~500 m^3/yr of waste with an average density of 160 to 240 kg/m^3 (10 to 15 lb/ft^3), none of which is released to the atmosphere. The routine radioactive releases from a nuclear power plant that go directly into the atmosphere are well below those that could pose any conceivable danger to humans. Certainly, the danger from these wastes is not necessarily proportional to their volume, but their management definitely depends on the volume to be disposed.

The management of radioactive waste is recognized as an integral part of nuclear activities and is treated as such. Unfortunately, until about 1980, there was no detailed government plan in the United States for the safe disposal of nuclear wastes. In the meantime, wastes kept accumulating as a result of military and civilian activities that started with the Manhattan Project in the 1940s. Fortunately, only minor mishaps took place during all these years, to the credit of the nuclear industry. At present, however, there are two congressional acts and the federal as well as state governments are proceeding to implement the agenda dictated by Congress on this matter.

Although Congress first acted only in 1980, the problem of nuclear waste management was recognized much earlier. Government, industry, and universities produced a plethora of reports and papers dealing with the subject[1-10] both here and in other countries. The general opinion of the technical community is that radioactive waste disposal is not a technical problem in the sense that new technology needs to be developed, and it is not a serious task, especially compared with the problems associated with wastes from other industries. Unfortunately, this opinion is not shared by the public. The perception by the public is that the disposal of radioactive waste poses a threat to their lives, a threat much more serious than many other chemical wastes that are handled routinely and to which people are exposed routinely. As a result of this public perception of the problem, the necessary political decisions for the establishment of disposal sites have not been made.

Since radioactive wastes can have detrimental effects, a management plan—such as the one provided by two recent congressional acts—must be implemented in order to protect people and the biosphere by isolating the wastes. Isolation of radioactive wastes may be accomplished by a combination of storage and disposal. Storage refers to long- or short-term isolation but with the possibility of retrieval. Disposal, on the other hand, means permanent isolation without possibility of retrieval.

The damaging potential of radioactive wastes is frequently given in terms of their activity, expressed in becquerels or curies, but a better measure of the degree of potential biological hazard, is their toxicity, defined as follows:

$$\text{toxicity}(i,k) = A(i)/\text{MPC}(i,k) \ , \qquad (1)$$

where

$\text{MPC}(i,k)$ = maximum permissible concentration (Bq/m^3 or $\mu Ci/cm^3$) of isotope i in air or water, as given in U.S. Nuclear Regulatory Commission (NRC) regulations 10CFR20 (Ref. 11) (in the proposed revision of 10CFR20, the MPC will be replaced by an equivalent quantity called derived air concentration).

k = air or water

$A(i)$ = activity of isotope i (Bq or μCi).

If more than one isotope is involved in a mixture, a summation is performed over all the isotopes present in that mixture. The toxicity represents the volume of air or water with which the radioactive wastes must be diluted so that continuous breathing of that air or drinking of that water will result in a dose that is less than the dose of 5 mSv/yr (500 mrem/yr) received from continuous exposure to a concentration equal to a MPC. Regulations that specify maximum allowed activity for various isotopes are based, in general, on equal toxicity of the isotopes involved. As an example, compare the toxicities of 3.7×10^{10} Bq (1 Ci) of the isotopes ^3H and ^{137}Cs. The MPC values in air are 1.11×10^5 Bq/m^3 (3.0×10^{-6} $\mu Ci/cm^3$) and 740 Bq/m^3 (2.0×10^{-8} $\mu Ci/cm^3$), respectively, for these two isotopes. According to Eq. (1):

$$\begin{aligned} \text{toxicity}\ (^3H) &= 3.7 \times 10^{10}\ Bq/1.11 \times 10^5\ Bq/m^3 \\ &= 3.3 \times 10^5\ m^3 \ , \\ \text{toxicity}\ (^{137}Cs) &= 3.7 \times 10^{10}\ Bq/740\ Bq/m^3 \\ &= 5.0 \times 10^7\ m^3 \ , \end{aligned}$$

and

$$\text{toxicity}\ (^{137}Cs)/\text{toxicity}\ (^3H) = 150 \ .$$

Therefore, for the same activity, ^{137}Cs is 150 times more hazardous than tritium.

Although there is no typical waste and, therefore, no generic composition of nuclear wastes, there are some isotopes that constitute the main hazard. These isotopes are listed in Table I. The last three isotopes of Table I are encountered in irradiated steel components. The others are produced in the reactor fuel either as fission products or as a result of neutron absorption and radioactive decay. In a mixture of radioisotopes in which many different half-lives are involved, the importance of isotopes, in terms of hazard, changes over time[3] (Fig. 1).

In the sections that follow, we discuss the sources and types of radioactive waste, the steps to be taken for their disposal in accordance with the recent congressional acts, and the pertinent rules and regulations promulgated by the federal government. The actions

TABLE I

The Most Important Isotopes Encountered in Radioactive Wastes

Isotope	Half-Life (yr)	Radiation of Primary Concern
^{90}Sr	29	Beta particles
^{129}I	1.6×10^7	Beta particles
^{135}Cs	2.0×10^6	Beta and gamma particles
^{137}Cs	30	Beta and gamma particles
^{99}Tc	2.0×10^5	Beta particles
^{229}Th	7 340	Alpha particles
^{230}Th	7.7×10^4	Alpha particles
^{237}Np	2.1×10^6	Alpha particles
^{239}Pu	24 000	Alpha particles
^{240}Pu	6 580	Alpha particles
^{241}Am	460	Alpha particles
^{243}Am	7 370	Alpha particles
^{243}Cm	32	Alpha particles
^{244}Cm	18	Alpha particles
^{59}Ni	8.0×10^4	X rays
^{94}Nb	2.0×10^4	Beta and gamma particles
^{60}Co	5.3	Gamma particles

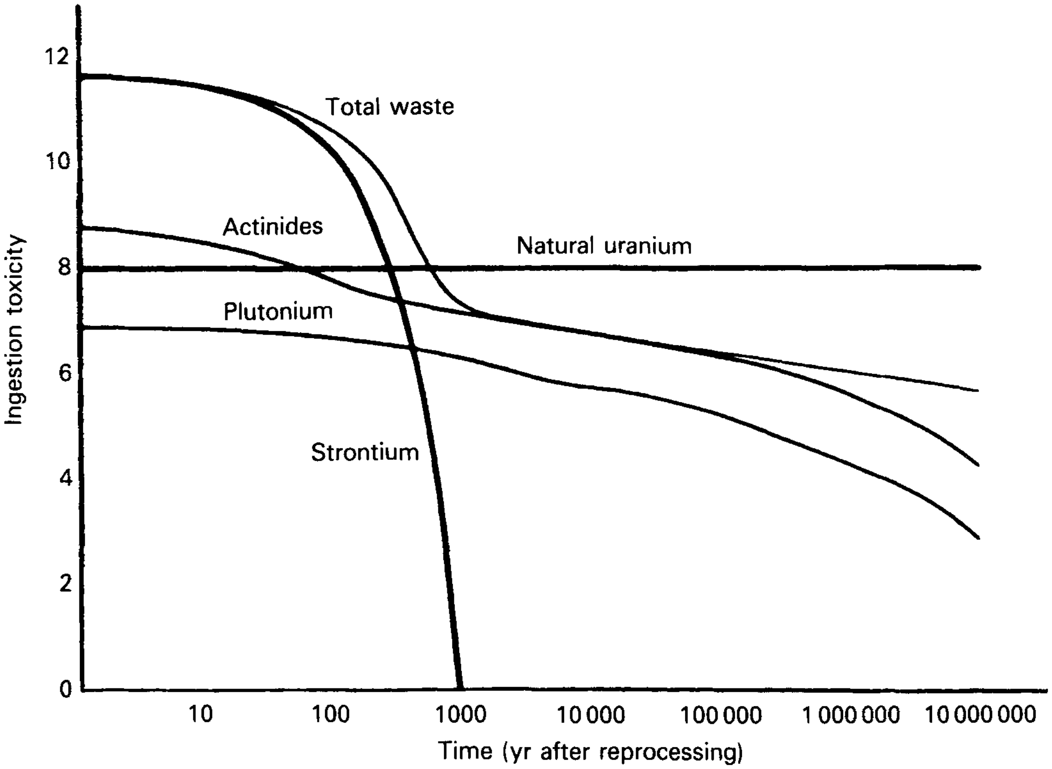


Fig. 1. Decay of radioactive waste over time. After ~2000 yr, the toxicity of the waste becomes less than the toxicity of natural uranium.

taken by other nations for the disposal of radioactive wastes are also presented.

II. CLASSIFICATION OF RADIOACTIVE WASTES

The following are the major categories of radioactive wastes, as classified in the United States.

1. *High-level wastes (HLW)*: They are defined as follows in the Nuclear Waste Policy Act[12,13] (NWPA): "(A) the highly radioactive material resulting from the reprocessing of spent fuel, including liquid waste produced directly in reprocessing, and any solid material derived from such liquid waste, that contains fission products in sufficient concentrations; and (B) other highly radioactive material that the Commission (NRC), consistent with existing law, determines by rule requires permanent isolation."

In a once-through fuel cycle, the spent nuclear fuel is considered to be HLW.

2. *Transuranic waste (TRU)*: This is defined as waste containing alpha-emitting isotopes with atomic number $Z > 92$, with half-lives >5 yr and concentration $>3.7 \times 10^6$ Bq/kg (100 nCi/g waste). This is the NRC definition. The Environmental Protection Agency (EPA) and U.S. Department of Energy (DOE) use a slightly different definition, with the same specific activity but half-lives >20 yr. TRU waste is a subset of HLW, that, although mentioned, is not formally defined in the NWPA.

3. *Low-level waste (LLW)*: This is defined as follows in the NWPA: "(A) is not HLW, spent nuclear fuel, transuranic waste, or by-product material as defined in section 11e(2) of the Atomic Energy Act of 1954; and (B) material the Commission (NRC), consistent with existing law, classifies as LLW."

4. Uranium mill tailings are also considered as radioactive wastes. Tailings are what is left behind after the uranium has been removed from the ore. Almost all the radium and the other uranium decay products become the radioactive constituents of the tailings.

In 1978, Congress passed the Uranium Mill Tailings Radiation Control Act, which has two principal features: (a) It directs the federal government to undertake the task of eliminating the hazards associated with several inactive tailings piles left from past uranium milling operations, and (b) the affected states are asked to pay 10% of the total cost involved. The act clarifies and strengthens the authority of the NRC to insist on proper tailings management by its uranium mills licensees. In 1982-1983, the EPA proposed standards relative to uranium mill tailings. In general, tailings are not transported from the point of their generation; they are disposed on-site. Mill tailings are not discussed further since their management seems to have been resolved.

5. Finally, there is one other category of waste not regulated by the NRC but only by the individual states. These are naturally occurring or accelerator-produced radioactive materials (NARM). In terms of disposal, most NARM qualifies as LLW.

In other countries, the classification of radioactive wastes may be slightly different. In France, wastes are divided into the following three categories:

1. LLW and medium-level short-lived beta/gamma waste (MLW)
2. MLW containing alpha emitters or TRU elements
3. wastes from reprocessing operations (i.e., same as U.S. HLW).

III. SOURCES AND EXAMPLES OF RADIOACTIVE WASTES

All activities involving radioactive materials, either civilian or military, produce radioactive wastes. Examples are nuclear fuel-related activities; reactor operations; nuclear weapons production; scientific research using radioisotopes; medical operations using radioisotopes for research, diagnosis, and therapy; production of radiopharmaceuticals; radioisotopic sources; and others.

Examples of LLW are items such as contaminated clothing, tools, swipes, liquids containing radioisotopes, resins, and activated materials. Examples of HLW are spent nuclear fuel and residues (liquid or solid) of reprocessing operations. TRU waste is produced primarily by defense-related nuclear activities.

Civilian wastes consist mainly of spent fuel and LLW. Defense wastes consist of spent fuel, HLW from reprocessing operations, TRU, and LLW. Research establishments generate HLW or TRU only in exceptional cases when they handle nuclear fuel or fissile material for research.

To provide an idea of the volume of wastes involved, Table II gives the lifetime waste generation by light water reactors (LWRs). The present inventory of radioactive wastes, through 1987, is shown in Fig. 2 (Ref. 14). The projected quantities for the year 2000 are as follows:

Spent fuel	40 200 t of heavy metal (HM)
HLW	3.3×10^5 m^3
LLW	6.3×10^6 m^3

In 1987, ~77 500 m^3 of LLW was disposed of in the United States.[15] The biggest producers[16] of LLW are South Carolina (12%), Massachusetts (10%), Illinois (9%), New York (8%), and California (7%).

TABLE II

Lifetime Radioactive Waste Generation from a PWR and a BWR*

Waste Type	Reference PWR, 1 GW(electric)		Reference BWR, 1 GW(electric)	
	Volume (m^3)	Radioactivity (undecayed curies)	Volume (m^3)	Radioactivity (undecayed curies)
Once-through fuel cycle wastes				
MIII tailings	4.353E+06[a]	3.710E+04	4.867E+06	4.149E+04
LLW from uranium conversion[b]	3.411E+02	9.813E+03	3.814E+02	1.097E+04
LLW from uranium enrichment[c]	1.328E+02	9.716E+03	1.365E+02	1.080E+04
LLW from fuel fabrication	3.063E+03	7.288E+00	4.110E+03	9.781E+00
LLW from reactor power generation	3.032E+04	2.866E+04	5.217E+04	7.956E+04
Reactor spent fuel	5.213E+02	3.270E+09[d]	6.996E+02	3.342E+09[d]
Decommissioning wastes				
LLW	1.510E+04	1.057E+05	1.640E+04	2.532E+05
Greater than class C wastes	1.130E+02	4.070E+06	4.070E+01	5.300E+06
Total	4.403E+06	3.274E+09	4.941E+06	3.348E+09

*Waste generated from 40 yr of reactor operation and 26 GW(electric)·yr of electric energy production.

[a] Read as 4.353×10^6

[b] Applies to the fluorination/fractionation process.

[c] Applied to the gaseous diffusion process.

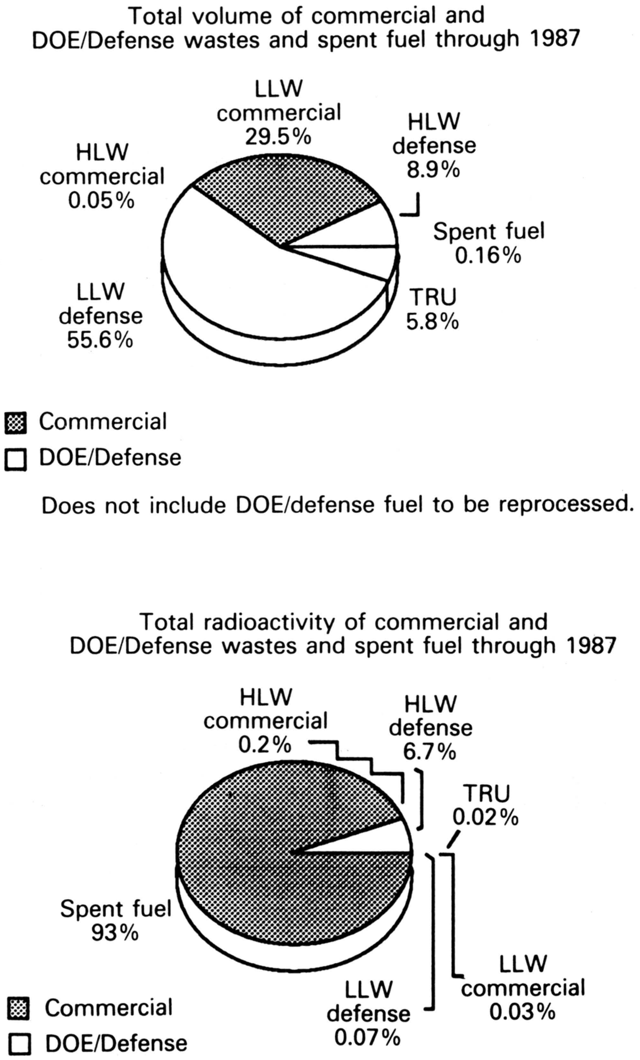
[d] Based on activity levels measured 1 yr after reactor discharge, as reported in Ref. 10. For the PWR, these levels are based on a burnup of 33 000 MWd/tonne HM. Activity levels for the BWR are based on a burnup of 27 500 MWd/tonne HM.

Waste	cm^3
HLW	
Commercial	2.32×10^3
Defense	3.79×10^5
Total HLW	3.81×10^5
LLW	
Commercial	1.28×10^6
Defense	2.38×10^6
Total LLW	3.64×10^6
TRU	2.50×10^5
Spent Fuel[a]	6.83×10^3
Total	4.28×10^6

[a] Includes spacing between fuel assembly rods.

Waste	Ci
HLW	
Commercial	2.97×10^7
Defense	1.27×10^9
Total HLW	1.30×10^9
LLW	
Commercial	4.96×10^6
Defense	1.40×10^7
Total LLW	1.90×10^7
TRU	4.12×10^6
Spent Fuel[a]	1.76×10^{10}
Total	1.89×10^{10}

[a] Does not include DOE/defense fuel to be reprocessed.



Does not include DOE/defense fuel to be reprocessed.

Fig. 2. Inventory of radioactive wastes in the United States as of the end of 1987.

IV. NWPA AND ITS AMENDMENTS

The NWPA was passed by Congress in December 1982, and was signed into law by the President on January 7, 1983 (Ref. 12). The act was amended in 1987 (NWPAA) and its amendments became law on December 22, 1987 (Ref. 13).

The NWPA is very comprehensive legislation that spells out detailed step-by-step procedures for completion of the various tasks related to the disposal of HLW. Specifically, Title I of the NWPA contains the following four subtitles:

1. Repositories for Disposal of HLW and Spent Nuclear Fuel: Details are given of the process to be followed for the selection of a site for a repository. Consultation and participation of states and Indian tribes is emphasized. Dates are given for the major milestones to be achieved.

2. Interim Storage Program: The need for interim storage for spent fuel will be investigated. Although the generators of the spent fuel (the nuclear utilities) are responsible for storage, the act recognizes the responsibility of the government to provide interim storage facilities, if needed. A storage capacity of no more than 1900 t will be considered.

3. Monitored Retrievable Storage (MRS): The possibility of the construction of a facility for the storage of HLW and spent fuel, with retrieval possible, will be addressed as an option to the geological repository.

4. Low-Level Radioactive Waste: This subsection deals only with financial arrangements for LLW site closure.

Title II of the act deals with research and development (R&D) needed for the successful completion of the goals of the NWPA.

Title III establishes (a) an Office of Civilian Radioactive Waste Management (OCRWM) to implement the provisions of the NWPA (the OCRWM director reports to the Secretary of Energy), and (b) a Nuclear Waste Fund, i.e., the mechanism of collecting money from the nuclear utilities to cover the cost of the NWPA implementation.

Within months after the signing of the act the OCRWM was established, and contracts between the DOE and the nuclear utilities were signed for the payment of the fee required by the NWPA. The DOE developed a standard contract that represents the official agreement between the parties involved, guaranteeing that the federal government will take spent fuel and dispose of it properly in exchange for the fee paid by the utilities. Utilities are responsible for the storage of the spent fuel prior to disposal.

The Nuclear Waste Fund is made up from a charge of 1 mill/kW·h of nuclear electricity generated after April 7, 1983, paid directly to the DOE by each utility operating a nuclear power plant. For spent fuel discharged before April 7, 1983, the DOE has established a one-time fee per kilogram of heavy metal in the spent fuel. This fee approximately corresponds to 1 mill/kW·h of electricity generated by that spent fuel. All fees will undergo annual reviews. As of 1990, no adjustment has been deemed necessary.

Based on present (1989) nuclear electricity generation, the government collects approximately $300 million/yr. The estimated revenue from the fuel discharged before April 7, 1983, is $2.3 billion. The DOE estimates the total cost of the HLW disposal to be about $25 to 35 billion in 1988 dollars.

The process for the selection of a site for the repository proceeded, in the beginning, along the lines prescribed by the NWPA. In February 1983, the DOE identified nine sites as potentially acceptable: one in Nevada, in a geologic medium called tuff; one in Washington in basalt; two in Texas in salt; two in Utah in salt; one in Louisiana in a salt dome; and two in Mississippi in salt domes. By December 1984, environmental assessments prepared for these locations narrowed the selection to three sites for recommendation to the President. These were the sites in Yucca Mountain, Nevada; the one in Deaf Smith County, Texas; and the one in Hanford, Washington. The DOE began organizing programs for characterization of all three sites. The final selection of the first site became a political argument in Congress, since none of the three states desired to be the host for the first repository. While the selection for the site of the first repository seemed impossible, another requirement of the act, namely the selection of a site for the second repository, became politically impossible to implement.

Locating a site for the MRS did not fare any better. The purpose of the MRS is not very clear from the text of the NWPA. The act speaks of "long-term storage of HLW or spent nuclear fuel in monitored retrievable storage facilities (as) an option for providing safe and reliable management of such waste or spent fuel." The vagueness in the purpose of the MRS was interpreted as a possibility that the MRS would end up being the repository!

In April 1985, the DOE identified three candidate sites for an MRS, all located in the state of Tennessee. The congressional delegation from Tennessee opposed the location of the MRS in its state. To avoid a threatened moratorium on NWPA activities, Congress passed amendments to the NWPA (NWPAA) in December 1987.

The NWPAA dictate the following:

1. The DOE was instructed to proceed with characterization of the Yucca Mountain site in Nevada as the site for the first repository, and stop within 90 days all activities on the other two sites.

2. If the Secretary of Energy determines the Yucca Mountain site to be unsuitable for development, he should (a) terminate all activities at that site, (b) notify Congress and the state of Nevada and indicate the reasons for such termination, (c) remove all HLW from the site, (d) take reasonable and necessary steps to return the site to its original condition, (e) suspend all future benefits payments with respect to that site (see below discussion on benefits), and (f) report to Congress no later than 6 months after such termination about further action to assure the safe disposal of HLW.

3. The DOE cannot proceed with activities for a second repository without congressional approval. The Secretary shall report to the President and to Congress after January 1, 2007, but no later than January 1, 2010, whether or not a second repository is needed.

4. The proposed sites for MRS in Tennessee were nullified. An MRS review commission was established, consisting of three members, appointed by the President *pro tempore* of the Senate and the Speaker of the House of Representatives, to evaluate the need for a MRS and make recommendations to Congress as to whether or not a MRS can meet the "needs for packaging and handling of spent nuclear fuel, improving the flexibility of the repository development schedule, and providing temporary storage for spent fuel accepted for disposal." In addition, the commission "shall compare such a facility to the alternative of at-reactor (AR) storage of spent nuclear fuel." A report should be submitted to Congress by June 1, 1989.

5. The Secretary of Energy may enter into a benefits agreement with the state of Nevada concerning the repository, or with another state or Indian tribe concerning an MRS. In addition to the benefits so negotiated, the Secretary shall make the following payments:

Event	MRS (millions of dollars)	Repository (millions of dollars)
Annual payment prior to first spent-fuel receipt	5	10
Upon first fuel receipt	10	20
Annual payment thereafter	10	20

6. The office of a nuclear waste negotiator is established within the Executive Office of the President. The negotiator shall be appointed by the President, by and with the advice and consent of the Senate. The negotiator shall find a state or Indian tribe willing to host a MRS at a technically qualified site. The office shall cease to exist no later than 30 days after December 22, 1992 (5 yr and 30 days after the enactment of the amendments).

7. A nuclear waste technical review board (NWTRB) is to be established, an independent body within the executive branch. The NWTRB will consist of 11 members, appointed by the President for a term of 4 yr. The President shall designate one member as chairman. The President will select the members from nominees recommended by the National Academy of Sciences. The board shall evaluate the activities of the DOE pertaining to the NWPA and its amendments and report its findings, not less than two times per year, to the Congress and to the Secretary.

8. An office of subseabed disposal research is established within the Office of Energy Research of the DOE. The Secretary shall establish a university-based subseabed consortium to investigate the feasibility of this method of disposal and shall report to Congress on the feasibility of this method for disposal of HLW.

9. The Secretary shall conduct a study of the use of dry casks for the storage of spent fuel at the sites of nuclear power plants. The Secretary shall report to Congress on this matter.

In early 1990, the status of two major provisions of NWPAA is as follows[17,18]: The DOE stopped all activities on the other two sites and concentrated on Yucca Mountain, as instructed by NWPAA. Also according to the act, DOE had to get permits from the state of Nevada for all the work to be performed at the site. Unfortunately, and contrary to the federal law, the Nevada legislature passed a law that forbids the disposal of any radioactive materials at Yucca Mountain and the state agencies returned all DOE permit applications without processing them. In December 1989, the state of Nevada filed a lawsuit against DOE asking the courts to order DOE to terminate all activities at the Yucca Mountain site. Nevada claims that since Congress has not overriden the state law disapproving the Yucca Mountain site, the DOE's permit applications are meaningless. In January 1990, the DOE filed a suit against the state of Nevada asking the courts to "stop the state from impeding the scientific study called for by Congress to determine whether Yucca Mountain would be a suitable site for . . . a repository."

It is difficult to estimate the time it will take for this court fight to be resolved. One certain effect is a delay in the schedule for all the repository-related activities. The DOE has already pushed the date of having an operational repository from the year 2003 to 2010.

With respect to the MRS, the NRC submitted its report to Congress in November 1989 and made the following major recommendations[19]:

1. Congress should authorize the construction of a federal emergency storage (FES) facility with a capacity limit of 2000 tonne uranium. The NRC believes, and the authors agree, that an FES will be in the national interest because of the expected delay in the construction of the repository. An FES will represent insurance in case of an emergency due to an accident or an unanticipated delay of the repository. The estimated cost for the FES is $300 to 400 million (in 1989 dollars).

2. Congress should authorize the construction of a user-funded interim storage (UFIS) facility with a capacity of 5000 tonne uranium. The UFIS would provide storage in addition to that of the FES. The NRC recommends that the UFIS be a user facility because it will be used only by those utilities that have not provided spent-fuel storage for the lifetime of the plant. The estimated cost of the UFIS is $500 to 600 million (in 1989 dollars).

3. The NRC recommends that Congress remove the linkage between the MRS and the repository. Section 115(d) of the NWPAA states that[13] "(1) construction of such facility (MRS) may not begin until the NRC has issued a license for the construction of a repository, (2) construction of such facility or acceptance of spent fuel or HLW shall be prohibited during such time as the repository license is revoked by the NRC or construction of the repository ceases . . ." Obviously, the NRC's recommendation is a sensible one because the MRS facility will become necessary, especially if the repository is further delayed.

4. Congress should reconsider the subject of interim storage by the year 2000 by taking into account the progress made for the repository by that time and the experience gained from the construction and operation of the two MRS facilities.

V. RADIOACTIVITY AND DECAY HEAT OF SPENT FUEL

When fuel is irradiated in a reactor, three general groups of radioisotopes are formed: (a) fission products, (b) actinides, and (c) activation products. More than 350 nuclides have been identified as fission products, many of them with very short half-lives. Tables III and IV (Ref. 20) show production of fission products and actinides in a typical pressurized water reactor (PWR) spent fuel. Obviously, the exact quantities depend on the irradiation history of the fuel and the time after discharge. The concentration of any isotope can be computed, in principle, using the following general differential equation[21]:

$$dNi/dt = \sum_{j=1}^{N} l_{ij}\lambda_j N_j + \phi \sum_{k=1}^{N} f_{ik}\sigma_k N_k - (\lambda_i + \phi\sigma_i + r_i)N_i + F_i | i = 1,N \ , \quad (2)$$

where

N_i = atom density of nuclide i

N = number of nuclides

l_{ij} = fraction of decays of nuclide j leading to the formation of species i

λ_i = radioactive decay constant of species i

ϕ = position and energy-averaged neutron flux

f_{ik} = fraction of neutron absorption by nuclide k leading to formation of species i

σ_k = average neutron absorption cross section of nuclide k

r_i = continuous removal rate of nuclide i from the system

F_i = continuous feed rate of nuclide i; if nuclide i is produced by fission, $F_i = Y_i\Sigma_f\phi$, where Y_i = fission yield of nuclide i.

Equation (2) is not linear if either the flux ϕ or any of the cross sections are functions of time. To the activity from fission products and actinides, one should add the activity of the nuclides produced by the

TABLE III

Radioactivity from Principal Fission Products in Irradiated Fuel*

Nuclide	Half-Life	Activity (1×10^6 Ci/yr) (at discharge)
^{3}H	12.4 yr	1.93×10^{-2}
^{85}Kr	10.76 yr	0.308
^{90}Sr	27.7 yr	2.11
^{95}Zr	65.5 days	37.3
^{106}Ru	368 days	14.8
^{125}Sb	2.71 yr	0.237
^{129}I	1.7×10^7 yr	1.01×10^{-6}
^{131}I	8.05 days	23.5
^{133}Xe	5.27 days	43.9
^{134}Cs	2.046 yr	6.7
^{137}Cs	30.0 yr	2.94
^{144}Ce	284 days	30.2
^{151}Sm	87 yr	3.41×10^{-2}
^{154}Eu	16 yr	0.191
^{155}Eu	1.81 yr	0.204

*Data taken from Ref. 20. The fuel is supposed to be from a 1000-MW(electric) PWR, initial enrichment 3.3%, operating for 3 yr with an 80% capacity factor.

TABLE IV

Radioactivity from Principal Actinides in Irradiated Fuel*

Nuclide	Half-Life	Ci/yr	kg/tonne U
^{234}U	2.47×10^5 yr	19.4	0.12
^{236}U	2.39×10^7 yr	7.22	4.18
^{237}Np	2.14×10^6 yr	14.4	0.75
^{236}Pu	2.85 yr	134	9.2×10^{-6}
^{238}Pu	86 yr	1.01×10^5	0.22
^{239}Pu	24 400 yr	8.82×10^3	5.28
^{240}Pu	6 580 yr	1.30×10^4	2.17
^{241}Pu	13.2 yr	2.81×10^6	1.02
^{242}Pu	3.79×10^5 yr	37.6	0.35
^{241}Am	458 yr	4.53×10^3	0.05
^{243}Am	7 950 yr	477	0.09
^{242}Cm	163 days	4.40×10^5	4.9×10^{-3}
^{244}Cm	17.6 yr	7.38×10^4	3.3×10^{-2}

*Data from Ref. 20. Fuel burnup is 33 000 MWd/tonne U in a 1000-MW(electric) PWR, 150 days after discharge.

activation of the zirconium in the cladding, as shown in Fig. 3, and of nickel and cobalt isotopes found in the steel used in control rods and in fuel assembly spacer grids.

The concentration and activity of fission products should be known for the following reasons:

1. The fuel design must accommodate the buildup of solid and gaseous fission products.
2. The primary waste handling and shielding must be designed to accommodate the fission products released to the coolant if a fuel rod fails.
3. Absorption by fission products must be taken into account in the reactivity requirements.

To calculate the total activity in spent fuel as a function of time, Eq. (2) must be solved for all the isotopes of interest and the results summed. Two well-known codes that perform that task are ORIGEN (Refs. 21 and 22) and CINDER (Ref. 23). Based on ORIGEN results, analytic fits have been developed to give the activity as a function of time after discharge.[24]

The knowledge of the decay heat from spent fuel and its decrease in time is extremely important in order to (a) adequately cool the core after a normal or abnormal shutdown, (b) handle the fuel during storage and transportation, and (c) safely dispose of the fuel in a repository. Because of its importance, the federal government and other organizations have supported both theoretical and experimental work, not only to obtain the value of the decay heat, but also to reduce the uncertainty of the result as much as possible. The results of these efforts have been reported in the literature.[25-37]

The decay heat has been measured either using radiometric or calorimetric techniques. Radiometric techniques are based on the measurement of the energy spectrum of the beta and gamma rays emitted by the fission products.[27] Calorimetric techniques[31,34] use a large calorimeter in which all the energy released by the fission products is absorbed and measured. Both techniques encounter difficulties in capturing all the gamma rays in the finite volume of the detector or the calorimeter and in ensuring that the gaseous fission products do not escape.

In practice, it is frequently necessary to know the energy released by fission products, after a certain reactor operating history, at a certain point in time after reactor shutdown. To fulfill this demand, semi-empirical formulas have been developed over the years based on theoretical and experimental results.[38,39]

The most accurate equations are those given by Schrock[36] and LaBauve et al.[37] The best available data are represented by the expression

$$f(t) = \sum_{i=1}^{M} \alpha_i \exp(-\lambda_i t) \text{ (MeV/s·fission)} , \qquad (3)$$

where the time t is in seconds after the fission takes place. LaBauve et al. give the constants α_i and λ_i in terms of six energy groups. If such detail is not needed, the data of Schrock are more useful. Schrock gives the function $f(t)$ in terms of 23 exponentials, presented in Table V, and also the function

$$F(t,T) = \sum_{i=1}^{23} \alpha_i \exp(-\lambda_i t) * [1 - \exp(-\lambda_i T)]$$
$$\text{(MeV/s)/(fission/s)} , \qquad (4)$$

where

T = time during which the reactor operated at a constant fission rate (s)

t = time after shutdown (s).

This function is valid in the absence of neutron capture in fission products. Values of $F(t,T)$ for T equal to infinity ($T = 1.0 \times 10^{13}$ s) are given in Table VI. These data were used for the 1978 American Nuclear Society revised 5.1 Standard. Figure 4 is a plot of the $F(t,\infty)$ for ^{235}U and ^{239}Pu.

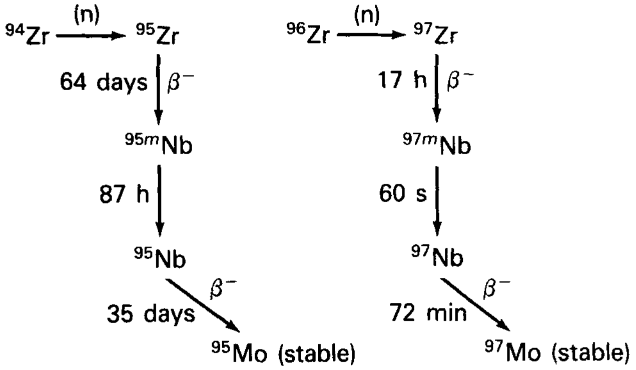


Fig. 3. Radioactive isotopes produced by the irradiation of the zirconium cladding.

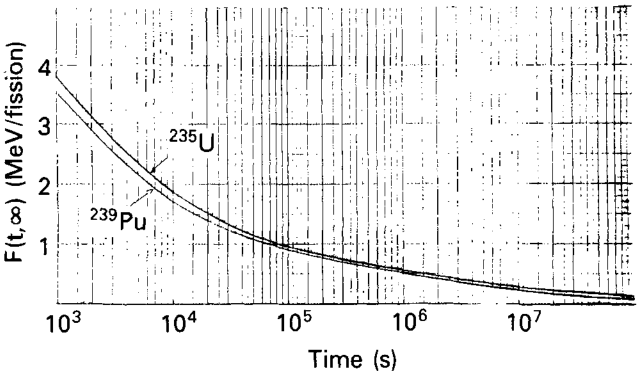


Fig. 4. Function $F(t,\infty)$ for the decay heat from ^{235}U and ^{239}Pu spent fuel.

TABLE V

Constants for the Exponential Representation of the Decay Heat Power Functions*

i	^{235}U Thermal		^{238}U Fast		^{239}Pu Thermal	
	α_i	λ_i	α_i	λ_i	α_i	λ_i
1	6.5057E−01[a]	2.2138E+01	1.2311E+00	3.2881E+00	2.0830E−01	1.0020E+01
2	5.1264E−01	5.1587E−01	1.1486E+00	9.3805E−01	3.8530E−01	6.4330E−01
3	2.4384E−01	1.9594E−01	7.0701E−01	3.7073E−01	2.2130E−01	2.1860E−01
4	1.3850E−01	1.0314E−01	2.5209E−01	1.1118E−01	9.4600E−02	1.0040E−01
5	5.5440E−02	3.3656E−02	7.1870E−02	3.6143E−02	3.5310E−02	3.7280E−02
6	2.2225E−02	1.1681E−02	2.8291E−02	1.3272E−02	2.2920E−02	1.4350E−02
7	3.3088E−03	3.5870E−03	6.8382E−03	5.0133E−03	3.9460E−03	4.5490E−03
8	9.3015E−04	1.3930E−03	1.2322E−03	1.3655E−03	1.3170E−03	1.3280E−03
9	8.0943E−04	6.2630E−04	6.8409E−04	5.5158E−04	7.0520E−04	5.3560E−04
10	1.9567E−04	1.8906E−04	1.6975E−04	1.7873E−04	1.4320E−04	1.7300E−04
11	3.2535E−05	5.4988E−05	2.4182E−05	4.9032E−05	1.7650E−05	4.8810E−05
12	7.5595E−06	2.0958E−05	6.6356E−06	1.7058E−05	7.3470E−06	2.0060E−05
13	2.5232E−06	1.0010E−05	1.0075E−06	7.0465E−06	1.7470E−06	8.3190E−06
14	4.9948E−07	2.5438E−06	4.9894E−07	2.3190E−06	5.4810E−07	2.3580E−06
15	1.8531E−07	6.6361E−07	1.6352E−07	6.4480E−07	1.6710E−07	6.4500E−07
16	2.6608E−08	1.2290E−07	2.3355E−08	1.2649E−07	2.1120E−08	1.2780E−07
17	2.2398E−09	2.7213E−08	2.8094E−09	2.5548E−08	2.9960E−09	2.4660E−08
18	8.1641E−12	4.3714E−09	3.6236E−11	8.4782E−09	5.1070E−11	9.3780E−09
19	8.7797E−11	7.5780E−10	6.4577E−11	7.5130E−10	5.7300E−11	7.4500E−10
20	2.5131E−14	2.4786E−10	4.4963E−14	2.4188E−10	4.1380E−14	2.4260E−10
21	3.2176E−16	2.2384E−13	3.6654E−16	2.2739E−13	1.0880E−15	2.2100E−13
22	4.5038E−17	2.4600E−14	5.6293E−17	9.0536E−14	2.4540E−17	2.6400E−14
23	7.4791E−17	1.5699E−14	7.1602E−17	5.6098E−15	7.5570E−17	1.3800E−14

*Taken from Ref. 36.

[a] Read as 6.5057 × 10^{-1}.

TABLE VI

Function* $F(t,\infty)$ for ^{235}U, ^{239}Pu, and ^{238}U

Time After Shutdown, t (s)	^{235}U		^{239}Pu		^{238}U	
	$F(t,\infty)$ (MeV/fission)	Uncertainty, 1σ (%)	$F(t,\infty)$ (MeV/fission)	Uncertainty, 1σ (%)	$F(t,\infty)$ (MeV/fission)	Uncertainty, 1σ (%)
1	12.31	3.3	10.27	5.6	14.19	12.0
10	9.494	2.0	8.243	4.2	10.29	9.5
10^2	6.198	1.8	5.685	4.2	6.217	5.9
10^3	3.796	1.8	3.516	4.4	3.598	4.9
10^4	1.908	1.7	1.727	4.8	1.777	4.4
10^5	0.9691	2.0	0.9421	5.0	0.9383	3.9
10^6	0.5509	2.0	0.5097	5.0	0.5171	3.9
10^7	0.2457	2.0	0.2282	5.0	0.2296	4.4
10^8	0.1165	2.0	0.08931	5.0	0.09280	5.0
10^9	0.05678	2.0	0.04195	5.0	0.04321	5.0

*Taken from Ref. 36.

In terms of reactor power, the decay heat power $P(t,\infty)$ is given by

$$P(t,\infty) = (P_o/Q) * F(t,\infty) \ , \tag{5}$$

where

P_o = constant reactor power during operation

Q = recoverable fission energy (MeV).

Values of Q for many common isotopes are given in Table VII.

A practical problem frequently encountered is to calculate the decay heat at a certain time after shutdown, with the reactor having operated at different powers for different finite periods. If a reactor operated for a finite time T at constant power P_o, the decay power at time t after shutdown is given by

$$P(t,T) = (P_o/Q) * [F(t,\infty) - F(t+T,\infty)] \ . \tag{6}$$

If the reactor operated for several periods at different power levels and more than one isotope is fissioned, Eq. (10) becomes a double summation over reactor operating periods and isotopes:

$$P(t,T_{tot}) = \sum_i \sum_n (P_{in}/Q_i) * [F_i(t_n,\infty) - F_i(t_n + T_n,\infty)] \ , \tag{7}$$

where P_{in} is the power produced by fissile isotope i during the n'th time period.

TABLE VII

Recoverable Energy per Fission for Many Common Isotopes*

Nuclide	Q (MeV)	Nuclide	Q (MeV)
^{232}Th		^{233}U	
Fast	197	Thermal	200
14 MeV	217	Fast	201
^{234}U		14 MeV	225
Fast	202	^{235}U	
^{236}U		Thermal	203
		Fast	203
Fast	205	14 MeV	228
^{238}U		^{237}Np	
Fast	209	Fast	204
14 MeV	230		
^{239}Pu		^{240}Pu	
Thermal	211	Fast	211
Fast	212	^{242}Pu	
14 MeV	238	Fast	216
^{241}Pu			
Thermal	213		
Fast	214		

*Taken from Ref. 37.

VI. SPENT-FUEL MANAGEMENT

Until 1977, when the U.S. government deferred reprocessing indefinitely, the utilities that owned nuclear power plants operated on the assumption that the spent irradiated fuel would be shipped to a federal facility a few years after its discharge from the core. Specifically, the nuclear power plants had provisions to store the spent fuel on-site, in water pools, for ~5 yr, with full core discharge capability.

After the government's decision against reprocessing was announced, the utilities were forced to expand their on-site storage capacity or find other means to store their fuel. Since spent reactor fuel had been successfully stored in water pools at many reactor sites, in some cases for >20 yr (Refs. 40 and 41), this "wet storage" method represented a sound demonstrated technology for the safe storage of irradiated fuel for decades, with negligible environmental impact. For this reason, the utilities that needed storage space sought to increase on-site spent-fuel pool capacity.

The original fuel storage racks in reactor spent-fuel pools were constructed of stainless steel and permitted 6 in. or more of water between fuel assemblies. To store more fuel safely, the racks were replaced with "high-density racks" constructed of Boral, a neutron-absorbing material (Boral is the trade name of a composite material consisting of boron carbide evenly dispersed within an aluminum matrix). The use of Boral permitted close spacing of fuel assemblies without criticality problems. Also, boron is unique in its ability to absorb thermal neutrons without producing any significant secondary radiation (only a soft 0.48-MeV gamma and alpha particles, easily absorbed by the water of the pool, are emitted).

A basic requirement for any spent-fuel storage facility is to have a value of $k_{eff} < 1.0$. This subcriticality should be achieved by an appropriate distance between assemblies and the use of solid poisons (e.g., Boral). Obtaining $k_{eff} < 1.0$ with dissolved boron in the water of the pool is not permitted.[42]

Because the shipment of spent fuel to a federal facility (MRS or a repository) is being pushed further and further into the future, many utilities began to face spent-fuel storage problems[43-45] in the early 1980s and difficulties[46,47] still persist today for some utilities. In cases where the on-site spent-fuel pool has been fully utilized, employing high-density racks, utilities may use one of the following options:

1. *Dry cask storage*: Dry cask storage is a method of storing the fuel in a canister, cooled by a gas and properly shielded.[47-50] The storage location may be on-site, designated as at-reactor (AR) storage, or off-site, in which case it is called away-from-reactor storage. In accordance with NWPAA, the DOE completed a study[47] of dry cask storage and submitted it to Congress in February 1989.

2. *Spent-fuel rod consolidation*: The fuel may be stored unconsolidated or consolidated. By definition, unconsolidated fuel means intact assemblies; consolidated means the fuel rods are removed from the fuel assembly and placed in a grid with closer spacing than that of an intact assembly, or the rods are placed in a close-packed array inside a canister.[45,48,51] Volume savings of 2:1 by consolidation have been demonstrated by many utilities.[45] The advantages of fuel rod consolidation are obvious: (a) storage capacity would be almost doubled and (b) the number of spent-fuel shipping casks can be halved. Consolidated fuel may be stored in a pool or in dry storage casks. The estimated costs of fuel consolidation are given in Table VIII. The cost of expanding pool capacity to store the consolidated fuel is estimated to be from \$39 to 75/kg HM in 1988 dollars, depending on the size of the expansion.

Disassembly of irradiated fuel produces radioactive scrap that has to be properly stored. When compacted, the scrap material from ten assemblies requires the space of about one canister. At the end of its life, the facility where the consolidation takes place will have to be decommissioned because of the radioactive contamination. The decommissioning cost, estimated to be \$100 000 in 1983 dollars, is included in the cost of running the business.

Many cask designs have been submitted to the NRC for licensing. As of the end of 1989, the NRC had granted licenses for metal casks and horizontal concrete modules. In licensing the casks, the NRC considers the stored fuel to have the composition of fresh fuel, i.e., no credit is given for burnup. Obviously, with credit for burnup, the spacing between assemblies could be further reduced, thus increasing the storage capacity of a given cask.

Figure 5 shows a typical metal cask for the dry storage of spent fuel. It is a cylindrical unit, 8 ft in diameter and 16 ft long, capable of storing 21 to 33 PWR assemblies (~9 to 14 t HM), or 45 to 70 boiling water reactor (BWR) assemblies (~8 to 12 t HM).

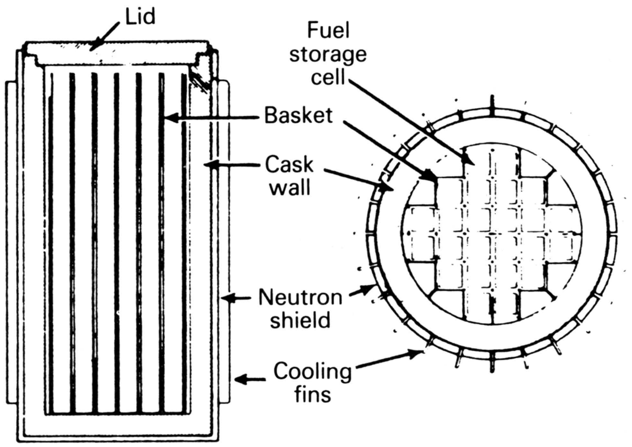


Fig. 5. Design of a typical metal cask for dry storage of spent fuel.

TABLE VIII

Estimated Cost of Consolidating Spent Fuel for In-Pool Storage*

Cost Item	Low Cost		High Cost	
	Fixed	Variable	Fixed	Variable
NRC licensing (\$)	200 000		500 000	
Equipment and installation	1 500 000		2 500 000	
Total fixed cost	1 700 000		3 000 000	
Operating labor (\$/kg HM)				
PWR fuel		5		7
BWR fuel		6		8
Cost per canister				
PWR fuel		2400		3600
BWR fuel		1800		2700
Total variable cost (\$/kg HM)				
PWR fuel		8		12
BWR fuel		12		15
Weighted average		9		14
Cost per kilogram of heavy metal gained				
PWR fuel		20		34
BWR fuel		34		60
Weighted average		25		43

*Taken from Ref. 47; in 1988 dollars.

Fully loaded the cask will weigh 100 to 120 t. The walls of the cask, made of iron or iron plus lead, are thick enough to provide adequate shielding against gamma rays. In addition, a neutron-absorbing material (e.g., polyethylene or resin) surrounds the circular surface of the cask. The interior of the cask contains fuel "baskets" where the fuel assemblies are placed. The baskets are made of various combinations of steel, aluminum, copper, and boron. Finally, the external surface of the cask may have fins to enhance cooling. The estimated cost of a metal cask is estimated to be from \$55 to 115/kg HM in 1988 dollars (Table IX). The storage pad area required will be ~25 ft^2/tonne HM for casks positioned vertically; for horizontal placement, the space required will be larger.

TABLE IX

Estimated Cost of Spent-Fuel Dry Storage Casks*

Type of Cask	Estimated Cost[a] (\$/kg HM)
Metal cask	55 to 115
Dual-purpose cask	72 to 122
Concrete cask (AR)	45 to 110
Horizontal concrete modules (e.g., NUHOMS)	44 to 85
Modular concrete vaults	75 to 165

*Taken from Ref. 47.
[a] In 1988 dollars.

A variation of the metal cask is a dual-purpose cask, which can be used for storage as well as for transportation of the fuel. Since this cask will be used for two purposes, it must satisfy the criteria of 10CFR72 (Ref. 42) as well as those for transportation of spent fuel, 10CFR71 (Ref. 52). The dual-purpose casks are very similar in shape and construction to the metal casks. The estimated cost of this cask is \$7.00/kg HM more than the cask made of metal.

Figures 6 and 7 show the conceptual design of two concrete storage casks. A concrete storage cask is very similar to a metal one, except that the body of the cask is now made of heavily reinforced concrete. A steel liner covers the inner surface of the concrete body of the cask. Two designs, ventilated and unventilated, have been proposed. In either case, the concrete wall is thick enough to provide adequate gamma-ray shielding. The baskets holding the fuel are very similar to those of the metal casks. An unventilated concrete cask is ~8.5 ft in diameter and 18 ft long. When loaded, such a cask will weigh ~90 t. A ventilated concrete

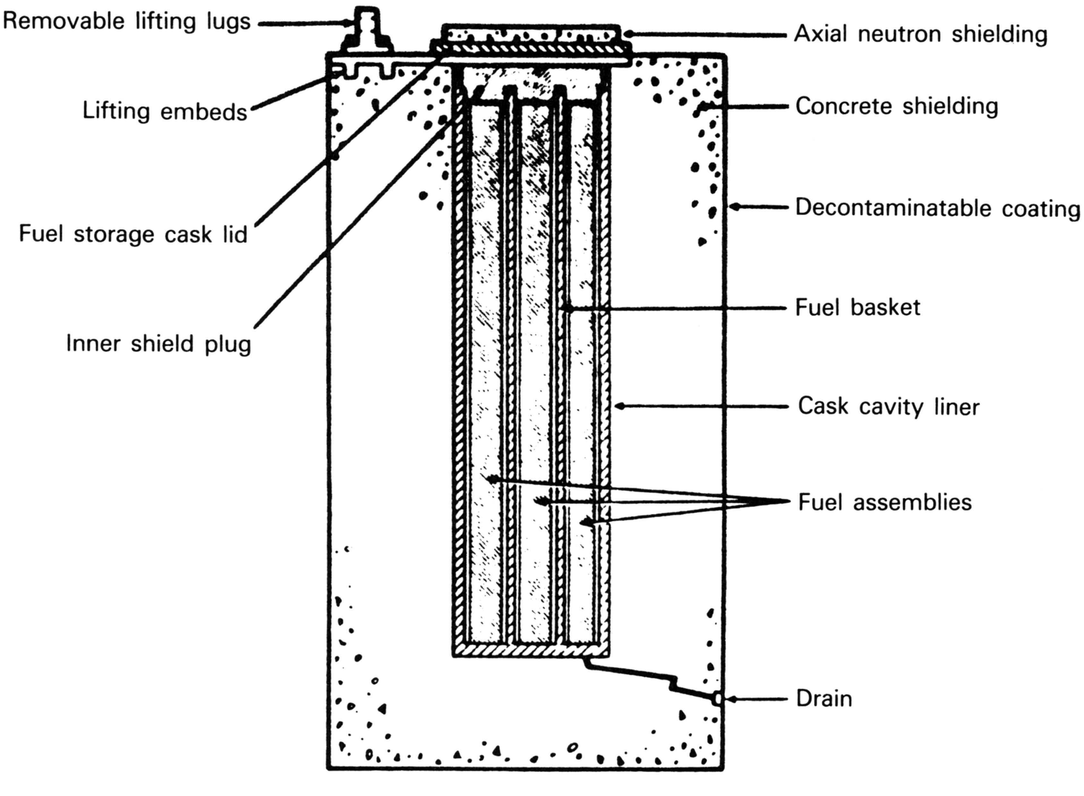


Fig. 6. Design of an unventilated concrete storage cask.

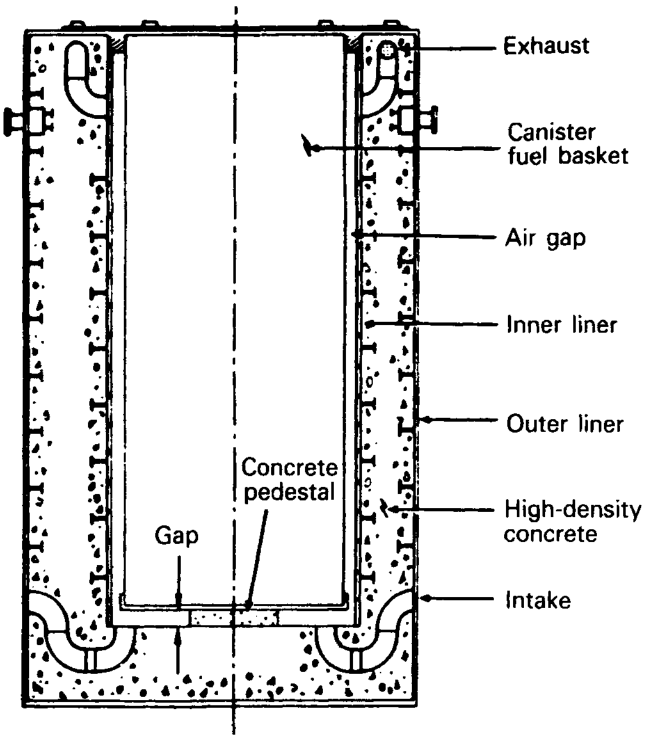


Fig. 7. Design of a ventilated concrete storage cask.

cask is ~12 ft in diameter and 20 ft long, and when loaded, it will weigh ~125 t (Ref. 47). The estimated cost of a concrete cask is from $45 to 110/kg HM in 1988 dollars for AR storage.

Several storage systems have been developed in the United States and in other countries.[47,53-55] The NUHOMS is a horizontal modular concrete storage system manufactured by NUTECH. The NUHOMS design will be used by the Duke Power Company.[48] It consists of a canister carrying 24 PWR irradiated assemblies in a helium atmosphere. Each canister is protected and shielded by a massive concrete module with 3-ft-thick walls. The decay heat is removed by thermal radiation, conduction, and convection from an air plenum inside the concrete module.

Another proposed storage method uses a modular concrete vault. In this design, the spent-fuel is kept in tubes made of carbon steel that are vertically stored in a concrete vault. Each tube will contain a single assembly and a module will have a capacity up to 200 spent-fuel assemblies. The fuel will be transported to the location of the vault using a standard transportation cask. There, a fuel-handling machine will remove the assemblies from the cask one by one, transport them into the vault, and place them into tubes. Once the assembly has been inserted into the tube, a shield plug is placed on the top of the tube. The estimated cost of this method of storage is from $75 to 165/kg HM in 1988 dollars.

According to NWPAA, the spent fuel will be disposed of in a geologic repository either consolidated or unconsolidated, but without being reprocessed. Each assembly will be enclosed in a sealed metal container, probably made of stainless steel, before it is placed in the repository.

Outside the United States, most countries plan to reprocess the spent fuel. The only exceptions are Sweden and Finland. In Sweden, no reprocessing takes place; therefore, the only HLW for disposal is spent fuel. The disposal will proceed as follows[56]:

1. Spent fuel is stored on-site for at least 6 months after discharge from the reactor.

2. The fuel is transported to a central intermediate storage facility and placed in a specially designed cask.

3. The fuel is stored in the intermediate facility for 40 yr.

4. The fuel is transported from the intermediate facility to an encapsulation station located above ground. The fuel encapsulation facility will be collocated with the repository.

5. The fuel is enclosed in canisters made of 10-cm-thick copper. Copper was chosen because it is not attacked by pure water but only by corrosive substances dissolved in water. Studies of corrosion in copper give the estimate that a 6-cm-thick wall thickness will not be penetrated by corrosion for at least 1 million yr.

6. The canisters are transferred to the final repository and are placed in holes at a depth of ~500 m (Fig. 8). The canisters in the holes are surrounded by a buffer material consisting of highly compacted bentonite clay.

The present plan calls for a repository that consists of parallel horizontal tunnels with vertical deposition holes. The distance between two neighboring holes will be ~6 m to ensure that the temperature anywhere in the repository will not exceed 100°C. This temperature limit ensures that the buffer material that surrounds the canisters (bentonite clay) does not become chemically active, thus affecting its performance in the long term.

In Finland, spent nuclear fuel will be disposed of in bedrock. A final disposal site will be selected by the year 2000 and will be operational by the year 2020. Treatment and placement of the spent-fuel assemblies will follow the Swedish approach.

VII. CHARACTERISTICS OF HLW

As defined in Sec. II, HLW includes (a) spent fuel, (b) by-products and remnants of the reprocessing operation, and (c) other materials that may be defined as HLW by the NRC.

Irradiated fuel still contains most of its original ^{238}U, about one-third of its original ^{235}U, almost all

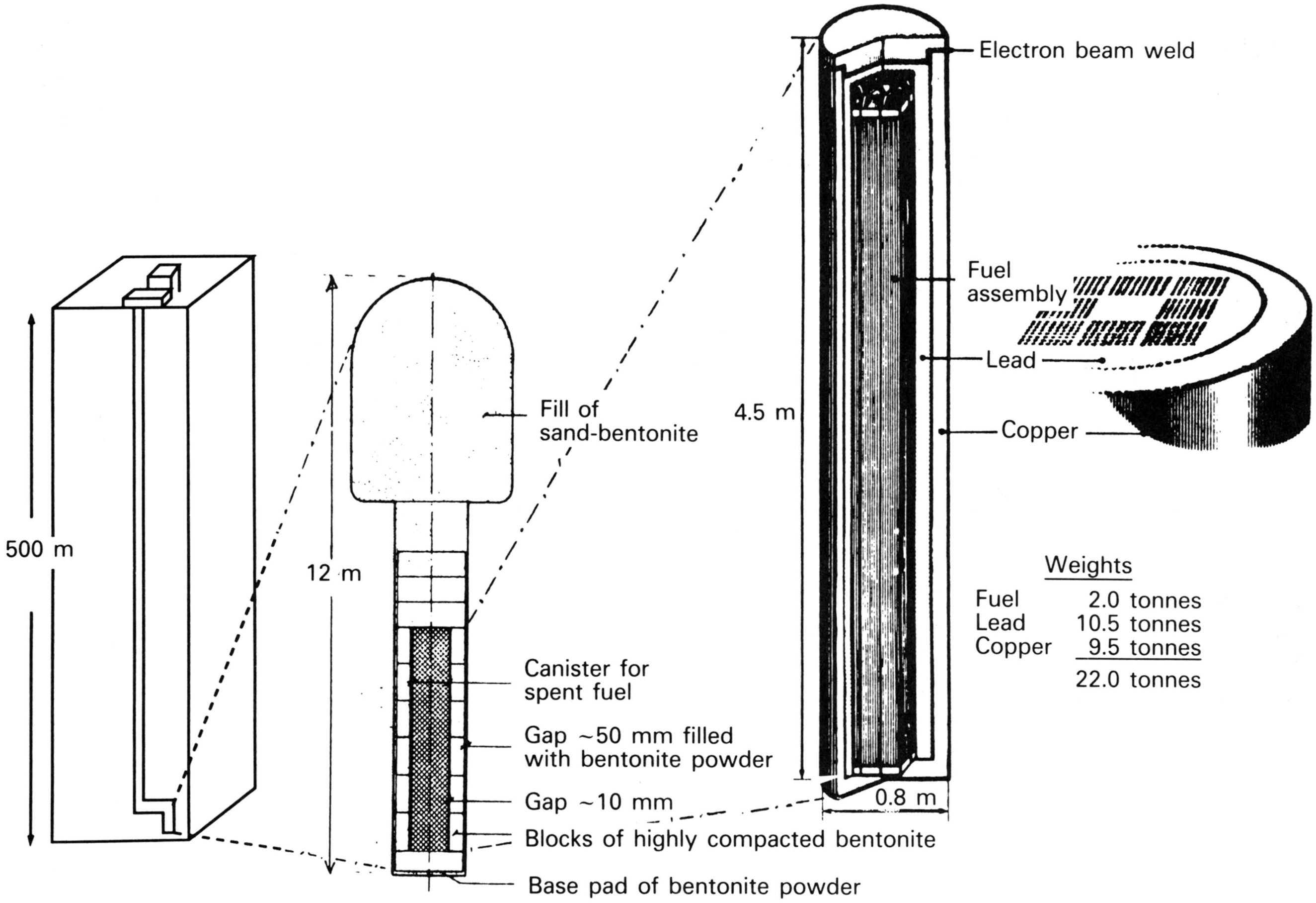


Fig. 8. Swedish concept for the geologic disposal of spent-fuel assemblies. The same principle can be used for vitrified or other LLW wastes. Only the canister will be different in the two cases.

the fission products, all the TRU isotopes, and many activation products. Figure 9 shows the isotopic change of 1000 kg of nuclear fuel that has produced power for 3 yr in a commercial LWR. Tables III and IV summarize the most important isotopic features of typical LWR spent fuels. Details about the estimated inventory of spent fuel from civilian LWRs in the United States are given in Ref. 14 (see also Fig. 2). The radioactivity and the thermal power of the spent fuel decays with time, as shown in Figs. 10 and 11.

The HLW generated after spent-fuel reprocessing contains >99% of the fission products (nonvolatile) and ~0.5% of the uranium and plutonium. As an example, the composition of HLW from the West Valley plant is given in Table X in terms of activity. Only a small fraction of this HLW comes from commercial civilian operations, because in the United States at present only the government reprocesses spent fuel, in connection with the weapons program and the refueling of nuclear navy vessels. Table XI gives the composition of liquid HLW from the reprocessing of LWR fuel, ready for solidification.[57]

The predominant thinking at this time for the treatment of liquid HLW is to solidify them before disposal (see Sec. IX). The spent fuel will be disposed of either as is, in assembly form, or it will be consolidated to reduce its volume.

VIII. CHARACTERISTICS OF TRU WASTES

As defined in Sec. II, TRU wastes are man-made isotopes with atomic number $Z > 92$ and half-life >5 yr, with a concentration >100 nCi/g (3.7×10^6 Bq/m^3) of waste. The United States is the only country that has defined this category of waste. In radioactive waste programs of other nations, TRU waste is considered HLW and is treated as such. For example, the liquids formed from the dissolution of spent fuel contain enough transuranics to satisfy the TRU waste definition. Yet, they are considerd HLW and not TRU waste. From the point of view of disposal, there is no essential difference between TRU waste and HLW.

The bulk of the TRU waste is generated by DOE activities related to defense programs, specifically plutonium separation processes. Examples of TRU are

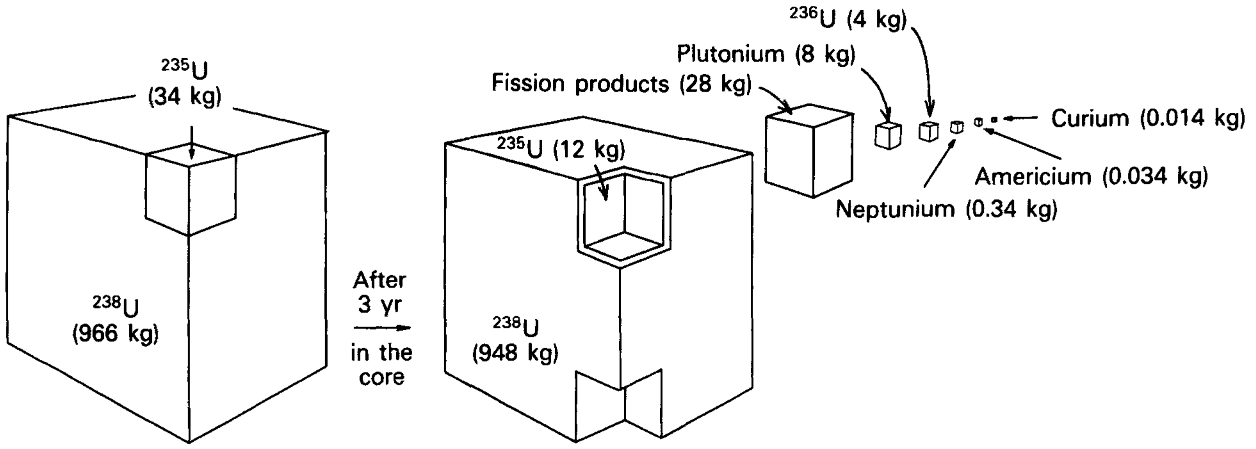


Fig. 9. Change in nuclear fuel as a result of fissions. The fresh fuel is UO_2. The irradiated fuel contains most of the original ^{238}U, about one-third of the original ^{235}U, and almost all the fission products and TRU elements.

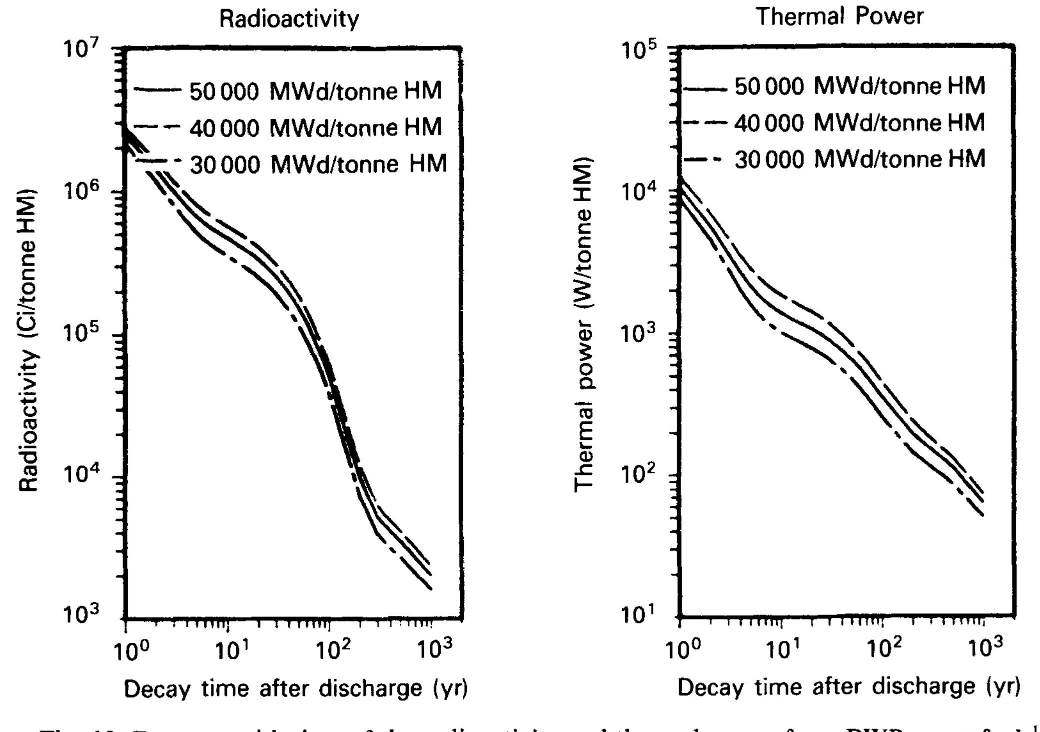


Fig. 10. Decrease with time of the radioactivity and thermal power from PWR spent fuel.[14]

contaminated glove boxes, filters, and tools, and chemical sludges produced by plutonium recovery streams. Other activities that may generate TRU waste are manufacturing of plutonium heat sources, spent-fuel reprocessing, nuclear fuel research activities, and decontamination and/or decommissioning operations. Since the composition of TRU depends on the source, it is difficult to give a truly representative composition. To give an idea, however, of the isotopes expected to be found in TRU, Table XII presents a list of the most important TRU isotopes.

Before 1970, all TRU waste was buried in shallow burial grounds (pits and trenches) covered with soil at government-owned as well as at commercial sites. In 1970, the U.S. governement decided that TRU waste required better confinement methods and started storing the TRU instead of burying it.

Most of the TRU waste can be handled with the shielding provided by the waste package, in which case it is called "contact-handled," and remote handling is not necessary. Some isotopes found in TRU waste emit energetic gamma rays and neutrons from spontaneous

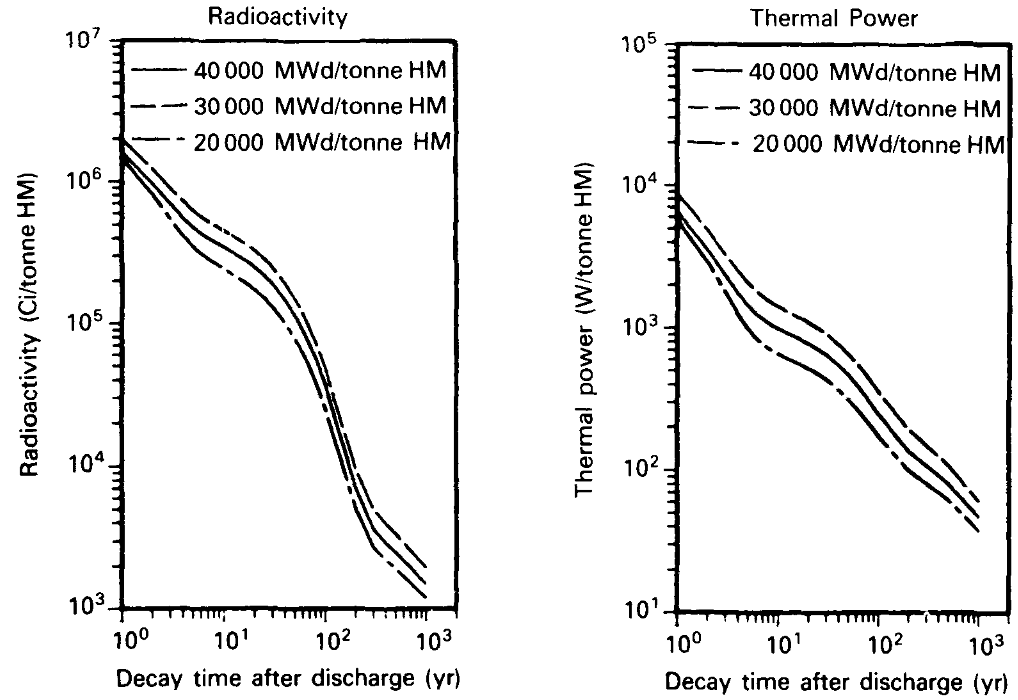


Fig. 11. Decrease with time of the radioactivity and thermal power from BWR spent fuel.[14]

fission, but only 1% of TRU waste emits enough gamma rays, neutrons, or beta particles to require special handling, in which case the package is called "remote-handled."

Transuranics are toxic because they are heavy chemical elements and emit alpha particles. Alpha-emitting isotopes are not harmful outside the body because the alpha particles do not penetrate the skin. In contrast, they are very harmful when inhaled and lodged in an organ of the body because they deliver a high dose to the surrounding body tissue. The elements found in TRU are heavy metals, and a relatively small fraction is absorbed through the digestive system if ingested. Thus, TRU waste is much less hazardous when ingested than when inhaled.

The handling, processing, and disposal of TRU waste are discussed in Secs. IX, X, and XI.

IX. SOLIDIFICATION OF HLW AND TRU

Since the present thinking is to dispose of liquid HLW and TRU waste in solid form, a method and a medium must be found for their solidification. Because there is no essential difference with respect to solidification and disposal of HLW and TRU waste, the material presented in this and the next three sections applies to both types of wastes. Any applications to one and not the other will be indicated. A comprehensive review of the recent advances in the solidification and disposal methods appears in Refs. 58, 59, and 60.

Three general methods of solidification have been considered:

1. *Calcination*: The waste is heated up to the point where it becomes completely dry. A powderlike substance remains after calcination.

2. *Cementation*: The liquid waste is mixed with a cement and poured into a container in which it is allowed to dry into a concrete block.

3. *Vitrification*: The waste is mixed with glass frit to form a solid glass containing the waste in solid solution. Vitrification is the preferred method because of the several advantages offered by glass.

In the United States, the slurry-fed ceramic melter (SFCM) process, developed at Battelle Pacific Northwest Laboratories, is the vitrification method used.[61,62] In the SFCM process, the waste slurry and the glass frit are continuously fed into the melter (Fig. 12) where the waste is mixed with the molten glass and then discharged into canisters where it solidifies. In France, the advanced vitrification Marcoule (AVM) process is used. In the AVM process, shown schematically in Fig. 13, vitrification is accomplished in two steps. First, the liquid waste is passed through an electrically heated rotating calciner where it is completely dried. The dried waste leaves the calciner tube by gravity through the lower end at a temperature of 300 to 400°C. It then enters the melting furnace where it is mixed with glass frit of the required composition to form the desired glass. In the furnace, the mixture is heated to the glass

TABLE X

Radioactive Inventory in an Average HLW Canister from the West Valley Plant* (1990)

Isotope	Activity (Ci)	Isotope	Activity (Ci)
^{3}H	0.0	^{147}Pm	2.45×10^{1}
^{14}C	0.0	^{151}Sm	3.07×10^{2}
^{55}Fe	0.192	^{152}Eu	8.62×10^{-1}
^{59}Ni	0.416	^{154}Eu	1.68×10^{2}
^{63}Ni	28.0	^{155}Eu	2.32×10^{1}
^{60}Co	0.814	^{232}Th	6.45×10^{-3}
^{79}Se	1.38×10^{-2}	^{233}U	3.55×10^{-2}
^{90}Sr	2.07×10^{4}	^{234}U	1.74×10^{-2}
^{90}Y	2.08×10^{4}	^{235}U	3.72×10^{-4}
^{93}Zr	1.07	^{236}U	1.09×10^{-3}
^{93m}Nb	8.37×10^{-1}	^{237}Np	9.25×10^{-2}
^{99}Tc	0.428	^{238}U	3.13×10^{-3}
^{106}Ru	5.79×10^{-5}	^{238}Pu	3.02×10^{1}
^{106}Rh	5.81×10^{-5}	^{239}Np	1.36
^{107}Pd	4.33×10^{-2}	^{239}Pu	6.39
^{125}Sb	2.34	^{240}Pu	4.2
^{125m}Te	0.573	^{241}Pu	1.96×10^{2}
^{126}Sn	0.409	^{241}Am	2.11×10^{2}
^{126m}Sb	5.73×10^{-2}	^{242}Pu	6.38×10^{-3}
^{126}Sb	5.73×10^{-2}	^{242}Am	1.11
^{129}I	0.0	^{242m}Am	1.11
^{134}Cs	7.03×10^{-1}	^{242}Cm	0.92
^{135}Cs	6.34×10^{-1}	^{243}Am	1.36
^{137}Cs	2.25×10^{4}	^{243}Cm	0.413
^{137m}Ba	2.13×10^{4}	^{244}Cm	20.5
^{144}Ce	3.48×10^{-7}	^{245}Cm	3.46×10^{-3}
^{144}Pr	3.49×10^{-7}	^{246}Cm	3.96×10^{-4}

*Used by permission of J. C. Pope.

TABLE XI

Composition of French Reprocessing Solution Obtained from LWR Fuel Irradiated to 33 MWd/kg*

Fission Products (g/ℓ)	
Selenium	0.08
Rubidium	0.53
Strontium	1.26
Yttrium	0.70
Zirconium	6.95
Molybdenum	5.04
Technetium	0.85
Ruthenium	1.58
Rhodium	0.44
Palladium	1.29
Silver	0.12
Cadmium	0.12
Tin	0.06
Antimony	0.01
Tellerium	0.71
Cesium	5.43
Barium	2.42
Lanthanum	1.82
Cerium	3.56
Praseodymium	1.68
Neodymium	6.07
Promethium	0.10
Samarium	1.21
Europium	0.20
Gadolinium	0.12
Total	42.35

Actinides (g/ℓ)	
Uranium	2.06
Neptunium	0.66
Plutonium	0.05
Americium	0.56
Curium	0.04
Total	3.37

Corrosion Products and Additives (g/ℓ)	
Sodium	14.71
Iron	9.08
Nickel	1.45
Chromium	1.54
Phosphorus	0.55
Total	27.33

Process Additives (g/ℓ)	
Aluminum	3.78

*Taken from Ref. 57.

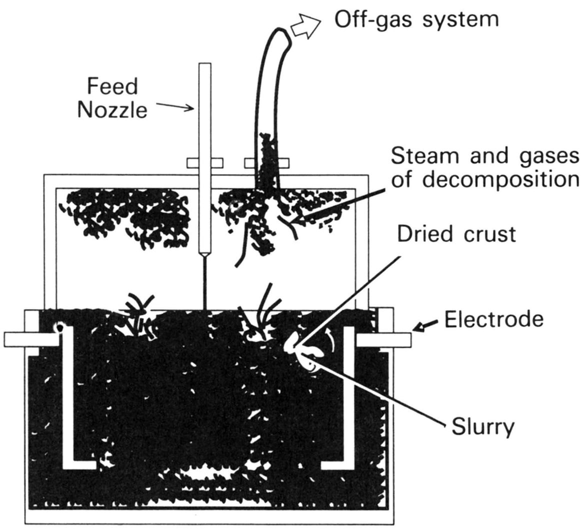


Fig. 12. Schematic of the SFCM vitrification method.

melting point ($\sim$1100°C) and the molten glass mixture is poured into a canister where it solidifies after it cools off.

The ideal containment material should (a) retain in insoluble form all the fission products and other chemical elements contained in the liquids from the spent-fuel reprocessing; (b) have suitable mechanical properties; (c) have stable properties over thousands of years; and (d) be relatively inexpensive and easy to handle.

Glass comes close to fulfilling all these requirements because glass (a) has a low leach rate and low solubility in water; (b) has high solubility for the nuclides found in HLW and TRU; (c) shows resistance to radiation damage; (d) requires moderate temperatures for preparation; and (e) is a material with which man has considerable experience over thousands of years. Many Egyptian, Greek, Chinese, and Roman artifacts that were made of glass or ceramic 3000 or more years

TABLE XII

The Most Important Isotopes Encountered in TRU Waste

Element	Atomic Weight	Atomic Number	Half-Life
Neptunium	237	93	2.14E6[a] yr
Plutonium	238	94	8.60E1 yr
	239		2.44E4 yr
	240		6.58E3 yr
	241		1.32E1 yr
	242		3.79E5 yr
Americium	241	95	4.58E2 yr
	242		1.60E1 h
	243		7.37E3 yr
Curium	242	96	3.91E3 h
	244		1.76E1 yr
	246		5.50E3 yr
Californium	252	98	2.65E0 yr

[a] Read as 2.14×10^6.

ago have been recovered intact after being buried or staying under seawater during all those years. It is reasonable to assume that glass that is specially designed and placed in an environment more benign than seawater will last at least that long.

Research on many possible forms of glass is still being conducted in the United States and in many other countries. It seems certain, at this time, that the first generation of vitrified waste will be based on a form of borosilicate glass, the typical composition of which is presented in Table XIII (Ref. 63). The general physical properties of glass waste forms are given in Table XIV. Borosilicate glass is preferred at this time, because it is a type of glass that is relatively easy to form and is well developed.

The second generation of waste form may be based on the SYNROC process. SYNROC is a titanate-based material that is still under investigation. SYNROC seems to offer, relative to other glasses, low leachability, excellent long-term stability, and increased waste loading per unit mass. More studies are needed, however, to prove these properties satisfactorily and also to increase experience for the production of SYNROC on a routine basis in large quantities.

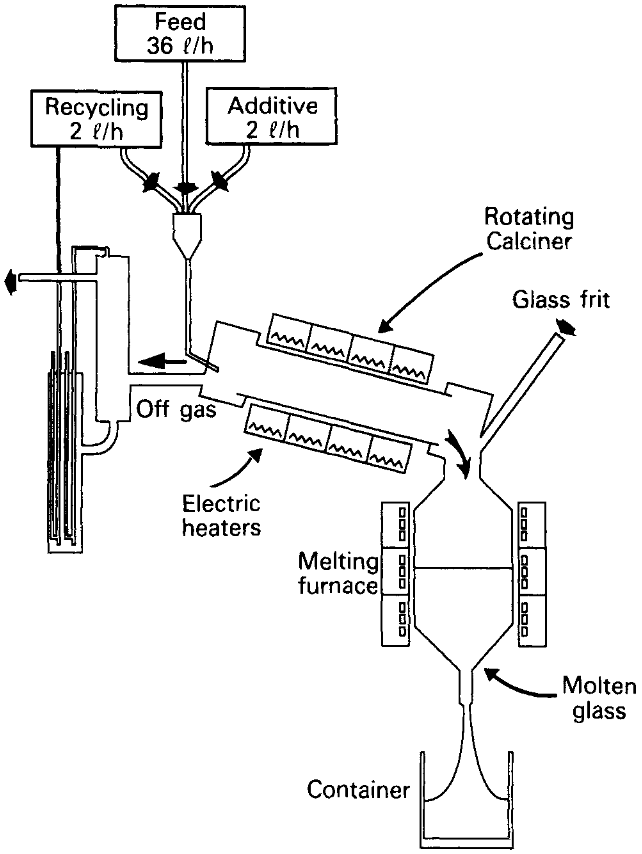


Fig. 13. Schematic of the French AVM method.

TABLE XIII

Typical Borosilicate Glass Composition*

Material	wt%
SiO_2	27.3
B_2O_3	11.1
Na_2O	4.0
K_2O	4.0
ZnO	21.3
CaO	1.5
MgO	1.5
SrO	1.5
BaO	1.5
Waste	26.3

*Taken from Ref. 63.

Several types of materials have been considered for the canister in which the vitrified waste is placed. Examples are Type 304L stainless steel, plated carbon steel, titanium, recycled contaminated steel, and lead. The most promising candidate today is Type 304L stainless steel. A conceptual waste package is shown in Fig. 14 (Ref. 64). It consists of the vitrified waste inside a "pour canister" and a second metal vessel, the "disposal container," enclosing the first. Each canister will contain ~1700 kg of glass, of which ~45 kg will be radionuclides. The decay heat from a canister will be from 200 to 500 W, with an estimated radiation level of ~13.9 mSv/s (5000 rem/h) on contact and ~1.7 mSv/s (600 rem/h) at 1 m. The reduction of the decay heat as a function of time is shown in Fig. 15.

TABLE XIV

Physical Properties of Glass Waste Forms

Property	Value
Thermal conductivity at 100°C [Btu/(h·ft·°F)]	0.55
Heat capacity at 100°C [cal/(g·°C)]	0.22
Fractional thermal expansion (per °C)	1.22×10^{-5}
Young's modulus (psi)	9.0×10^9
Tensile strength (psi)	9.0×10^3
Compressive strength (psi)	1.0×10^5
Poisson's ratio	0.2
Density at 100°C (g/cm^3)	2.5 to 3.0
Softening point (°C)	502

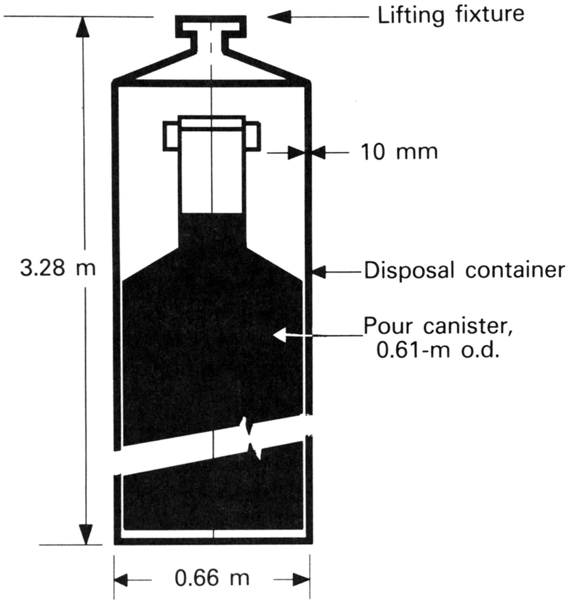


Fig. 14. Conceptual design for a HLW waste package.[64]

The canisters used by the French are smaller in size. They have a height of ~1.3 m, an outside diameter of 0.43 m, and a 5-mm-thick refractory stainless steel wall. Each canister will contain ~400 kg of vitrified glass.

At present, there are several plants operating or under construction, both in the United States and in other countries, that are designed to produce borosilicate glass. In the United States there are two operating plants, one at Savannah River and the other at West Valley, and both plants are to be used for the treatment of defense HLW.

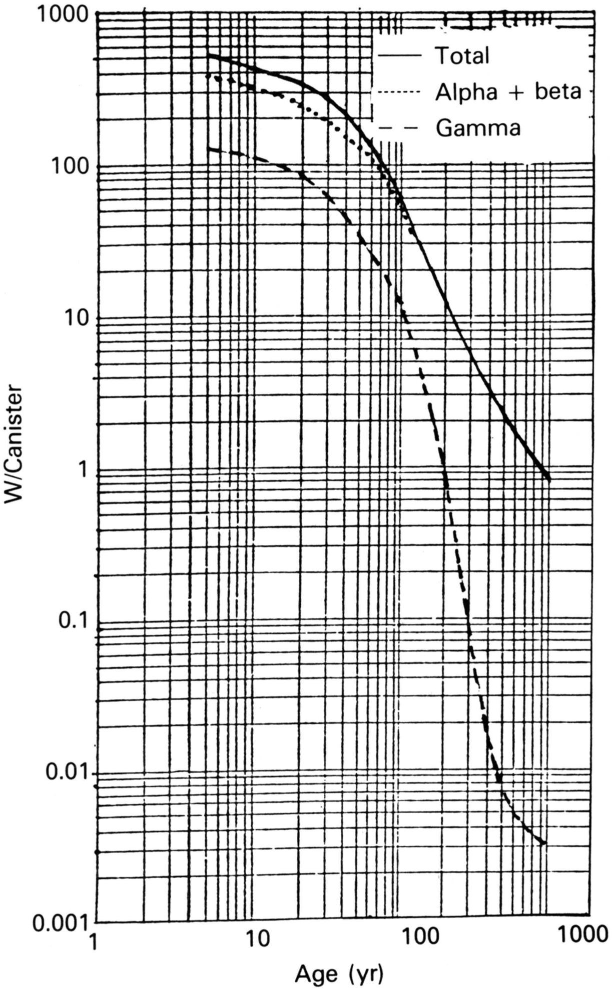


Fig. 15. Decrease with time of the heat generated by a glass waste.

X. HLW DISPOSAL METHODS

Whichever method is chosen for the disposal of the radioactive wastes, it has to satisfy one fundamental objective: the protection of the environment not only for the benefit of the generations that produce the wastes, but also for all future generations. The term "protection" should not be interpreted as meaning that the disposal will carry no risk: There is no human activity without some degree of risk. Instead, protection means that the method of disposal will result in a risk acceptable to society, acceptable in a risk-benefit sense.

Over the years, many disposal methods have been considered.[1-4,65] Examples are burial in the ground, following various procedures and using several depths; transmutation, i.e., nuclear transmutation by bombarding the wastes with particles in an accelerator or a fission or fusion reactor; ice sheet disposal (under the arctic ice cap); firing them into space (using a rocket and directing it either toward the sun or the outer space); and subseabed disposal. Most of these ideas have been abandoned because of high cost or high risks, or both. At present, the only method seriously considered is burial in a stable geologic medium. However, since the NWPAA (Ref. 13) directs the government to study disposal at sea, a few comments on this possible method are warranted.

Subseabed disposal is based on the fact that there are places on the ocean floor, at depths of ~5 miles, where tectonic plates meet and form a deep ditch. Materials placed in such places would tend to drift between the plates and downward, toward the center of the earth. The idea certainly has merit, but more research is needed.

The method of deep geologic disposal constitutes a multiple-barrier approach. The wastes are placed in a stable geologic formation, deep underground, and the barriers are the waste package, the disposal container, and finally the geologic medium. Media considered include salt, tuff, basalt, and granite. Although none of these media is perfect, any one of them could become the host for a satisfactory repository. Briefly, the major properties of these materials are these:

1. The salt material, either bedded or domed, exhibits high plasticity, which means that any cracks created for any reason will tend to close up on their own. At the same time, salt is highly soluble in water, but in places where one finds large deposits of salt, water has to be absent. Finally, salt is plentiful in the continental United States, as some 256 million acres are known to have the required deposit size.

2. Tuff is a volcanic material: volcanic ash that has been compressed under its own weight. Tuff may range from soft and relatively loose to tightly compacted. Large deposits of this rock are found in the western states, especially in arid regions.

3. Basalt is also volcanic material. It has moved upward through fissures in the earth's crust and flooded large areas in sheetlike formations. Deposits of basalt are usually extensive. The total deposit is made up of many sheets of lava, each from a different volcanic eruption. Basalt is very hard, dense, and strong, and it contains a negligible amount of moisture.

4. Granite and related rock are very abundant in the upper 5 miles of the earth's crust, formed as the result of molten material that has cooled. The most attractive property of granite is its strength, as a result of which granite preserves the opening of shafts, tunnels, and chambers during mining operations. Granite contains very little water and the minerals in this rock tend to reduce corrosion of metals and to retard the movement of waste. Some granitic formations are known to have been stable for as long as 2.5 billion yr.

Zeroing in on a site from a national survey is accomplished by taking into account geologic, environmental, and land use criteria. The following typical geological criteria are considered:

1. type of rock
2. geologic stability
3. seismic activity
4. depth and lateral extent of the rock
5. groundwater flow
6. presence of or proximity to active faults
7. surface water bodies
8. surface terrain characteristics.

The environmental criteria are taken into account by preparing an environmental impact statement in accordance with the National Environmental Policy Act (NEPA) of 1969. The land-use criteria are based on questions such as the following:

1. Is the land already committed to nuclear activities?
2. What are the population patterns in the area?
3. What is the population density?
4. What are other potential uses of the land (agriculture, mining, oil and gas exploration, etc.)?

On the surface, a geologic repository will look like a mine (Fig. 16). Underneath, at a depth of between 600 to 1200 m, the HLW will be placed in excavated tunnels called "waste emplacement rooms." The waste package, either spent fuel or vitrified HLW, will be deposited in these rooms or in holes drilled in the rock. For a period of ~50 yr, the wastes could be retrieved. After the operation of the repository has been confirmed to perform as designed, the rooms will be backfilled with the excavated rock. The backfilling provides an extra barrier between the wastes and the environment, enhances the heat transfer from the waste to the surrounding rock, and also reinforces the structural integrity of the mined geologic medium.

In terms of risks to the population from the repository, the proposed EPA rule (40CFR191) has set as an objective an individual lifetime risk of 1 in a 1 000 000 and an individual maximum lifetime risk of 1 in 10 000. This goal will be satisfied if no person receives a >0.10 mSv/yr (10 mrem/yr) effective dose equivalent.

As discussed in Sec. IV, the NWPAA designated the Yucca Mountain site in Nevada as the site that will be characterized for the first U.S. repository. The geologic medium of Yucca Mountain is tuff. The application by DOE requests the withdrawal of ~17 million m^2 (4255 acres) for a repository that will accept 70 000 tonne uranium of spent fuel or its equivalent in

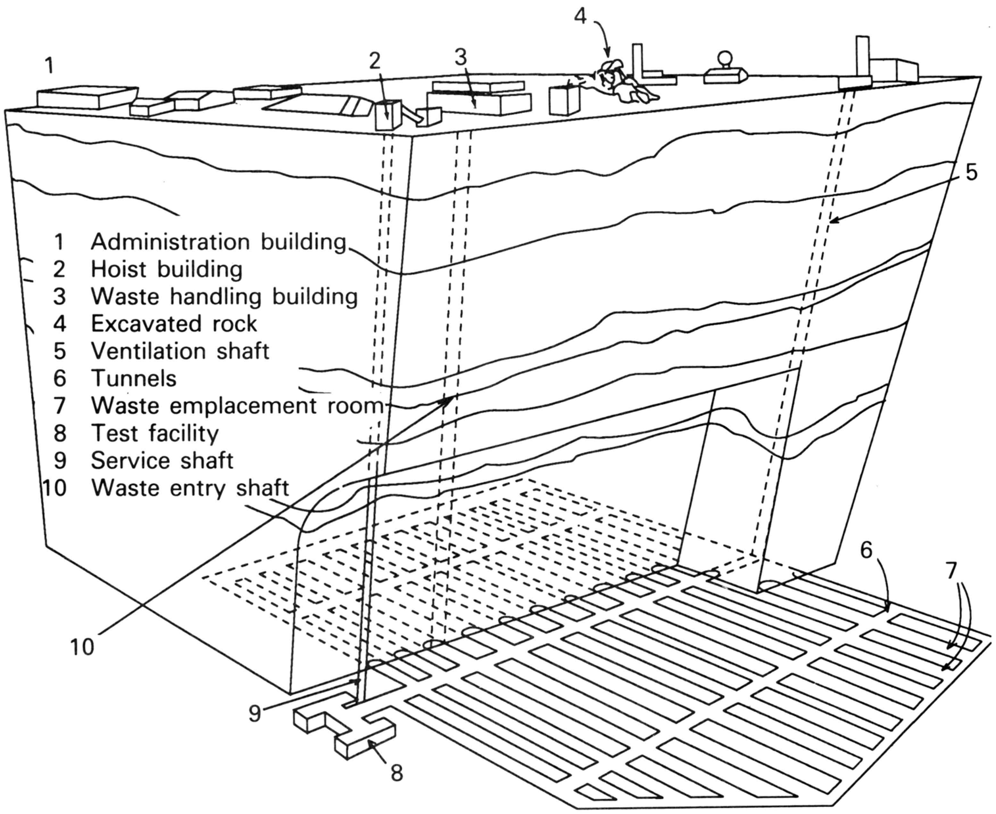


Fig. 16. Conceptual diagram of a geological repository. The surface buildings look almost like those of a common underground mine. The wastes will be placed in excavated tunnels at a depth of 600 to 1200 m.

terms of HLW. A schematic cross section of the repository is shown in Fig. 17.

Despite all these careful measures that are taken for the disposal of radioactive wastes, the critics say that there is no experimental evidence to support a process that is supposed to contain the wastes for thousands of years. That is not quite true. Nature has provided us with some extraordinary evidence.

In Gabon in West Africa, in a place called Oklo, there is sound evidence that a nuclear fission reactor

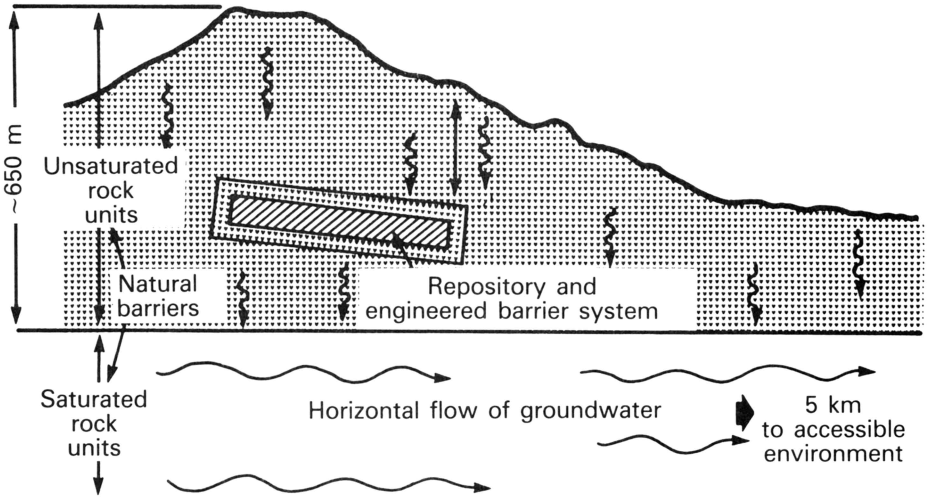


Fig. 17. Conceptual cross section of the Yucca Mountain repository.

started up on its own ~1.8 billion yr ago. It operated over several hundred thousand years and then shut down again on its own. This event was possible because at that time the abundance of the isotope ^{235}U in uranium was ~3% of the natural uranium. Apparently, the combination of uranium ore and groundwater were such that this particular area had the same composition as that of the fuel and moderator in present-day LWRs. This "natural" reactor produced ~10 t of HLW before it shut itself down, probably because of fuel "depletion." It has been determined that most of the solid fission products and all the TRU elements hardly moved but remained there, locked in the ore, where they decayed. The geologic medium at Oklo is clay. Surely, a repository designed with additional barriers should perform at least as well as random disposal in clay, without any extra precautions for the isolation of the wastes.

XI. THE WASTE ISOLATION PILOT PLANT

The Waste Isolation Pilot Plant (WIPP) is a project undertaken by the DOE, after Congressional authorization, for the purpose of providing an R&D facility to demonstrate the safe disposal of radioactive wastes resulting from the defense activities of the United States. The WIPP, exempted from NRC regulation, was ready for operation in 1988, as scheduled. However, new questions were raised about the expected performance of WIPP and no radioactive wastes have been placed there. The Secretary of Energy announced that the decision about opening WIPP will be made in June 1990.

The WIPP facility is located 25 miles south of Carlsbad, New Mexico, in an area of 10 240 acres, of which 30 acres constitute the primary zone. It looks very much like the planned geologic repository. The waste will be placed at a depth of ~660 m (2150 ft) in bedded salt. Consistent with its mission as an R&D facility, WIPP will be the site of many experiments involving the disposal of both TRU and HLW, although the primary objective is the study of TRU disposal. Some examples of programs planned are as follows:

1. The durability of the drum carrying the wastes will be studied.

2. The effectiveness of the backfill will be evaluated.

3. Data obtained directly from the disposal medium will help validate rock mechanics codes.

4. Thermal-structural experiments (*in situ*) will provide direct experimental data.

5. Simulated HLW packages will provide information regarding unexpected problems encountered with this method of disposal.

Waste packages sent to WIPP must satisfy many requirements, the most important being the following:

1. Waste containers must be noncombustible, weigh <11 300 kg (25 000 lb), must not be larger than $3.7 \times 2.4 \times 2.6$ m^3 ($12 \times 8 \times 8.5$ ft^3), and their thermal power must not exceed 3.5 W/m^3 (0.1 W/ft^3).

2. The surface dose rate should not exceed 0.56 Sv/s (200 mrem/h) at any point with a neutron contribution of >0.14 Sv/s (50 mrem/h) reported in the data describing the package.

3. Nuclear criticality conditions should be avoided by keeping concentrations of fissile isotopes below certain limits (200 g/55-gal drum; 100 g/30-gal drum; 5 g/ft^3 in boxes up to a maximum of 350 g).

4. Each package should provide the total alpha activity, a description of the waste form, and the mass of the organic material contained, for control of gas generation.

During the pilot phase of the project, all wastes used for the experiments may be retrieved. After the review of the pilot phase, if the facility performs as planned, the operations may be expanded to permit permanent disposal of TRU waste and HLW.

XII. HLW PROGRAMS OUTSIDE THE UNITED STATES

Activities related to the disposal of HLW are taking place in all countries with nuclear activities. There is exchange of information among all the countries, including the United States, and certain international projects have been undertaken. A prime example is the STRIPA project sponsored by Canada, Finland, France, Japan, Sweden, Switzerland, the United Kingdom, and the United States,[66] named after the STRIPA mine in central Sweden. The objective of STRIPA was to gain experience and information by carrying out *in situ* tests at depths similar to those in an actual repository. A review of the most important programs, as reported in the open literature, follows.

In France,[67] where spent nuclear fuel is reprocessed routinely, not only for the needs of the breeder reactor program but also for LWR recycling activities, well-developed procedures and methods for the treatment of HLW have been established. For an interim period of 1 to 5 yr, the liquid HLW is stored in refrigerated double-walled stainless steel tanks with an estimated life of 50 yr. After this temporary storage, the HLW is vitrified. The first pilot facility, named PIVER, was started at Marcoule in 1969. It used a batch process and produced 12 t of glass until 1973. In view of the excellent results obtained, an industrial-size facility was built, based on a continuous process known as AVM and it started operating in 1978 (Fig. 13). The AVM was designed[60] to process 60 ℓ/h of liquid HLW and produce 25 kg/h of waste-filled glass. The glass is solidified in containers made of 5-mm-thick stainless steel, with a diameter of 0.43 m and height of 1.3 m, each filled with ~400 kg of glass. The containers are closed, decontaminated, and sent to an intermediate storage facility located at Marcoule. This facility occupies an area of 400 m^2 and consists of 10-m-deep reinforced concrete shafts, cooled by forced air. Each shaft accommodates ten containers.

The intermediate storage of the glass is necessary primarily to allow for a considerable decrease of the surface temperatures of the canisters before they are sent to the geologic repository. Studies are in progress in France to determine the best geologic formation for the construction of a repository. The types of media considered are salt, granite, and clay.

In the United Kingdom, no plans have been announced for the disposal of HLW. The present policy is to store the HLW at ground level for at least 50 yr to allow cooling. By the end of that period, the government will have enough information for a proposed solution to the problem.

In Japan, there are definite plans for reprocessing of spent fuel (to start in 1997) and vitrification of HLW. Construction of the vitrification facility began in 1988.

XIII. SHIPPING CASKS FOR SPENT FUEL

Much experience with shipping casks has been developed in the Unites States. More than 5000 spent-fuel elements have been shipped over the past 20 yr without any radiation release, despite a few serious road accidents. This excellent safety record is due to the design characteristics of the casks used for the transportation of spent fuel.

Various cask models have been designed, tested, and placed into use. Lighter casks, from 25 to 40 t, are designed to hold one to seven fuel assemblies and to be carried by truck. Heavier casks, up to 120 t, may carry up to 36 or more assemblies and be transported by rail. In general, the casks are cylindrical in shape with multiple walls for structural integrity and provide protection against gamma and neutron radiation emitted by the spent fuel. In one such design, shown in Fig. 18, the fuel assemblies are sealed in a water-filled stainless steel cylinder with 0.5-in.-thick walls clad with 4 in. of a heavy metal, usually lead, for radiation shielding. This container is surrounded by 5 in. of water and encircled by a corrugated stainless steel outer package. Another cask, designed by the Babcock & Wilcox (B&W) Company, designated as BR-100, is shown in Fig. 19. This cask is to be carried by rail or barge.

Before any cask is allowed to be used for transportation of spent fuel, it must be certified by the NRC. Certification is awarded after the cask satisfies the set of standards shown in Fig. 20. Proof that the cask is certifiable is obtained by performing a series of tests using the actual casks filled with a material simulating fuel in terms of density and weight.

The Transportation Technology Center at the Sandia National Laboratories (SNL) has conducted a series

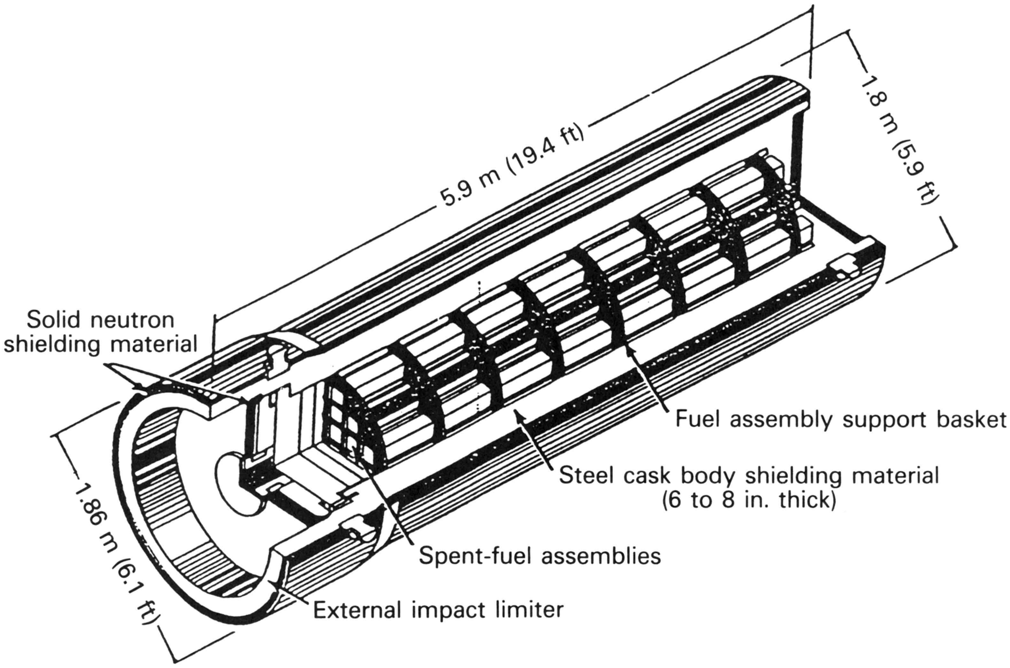


Fig. 18. Conceptual drawing of a 76-t rail cask.

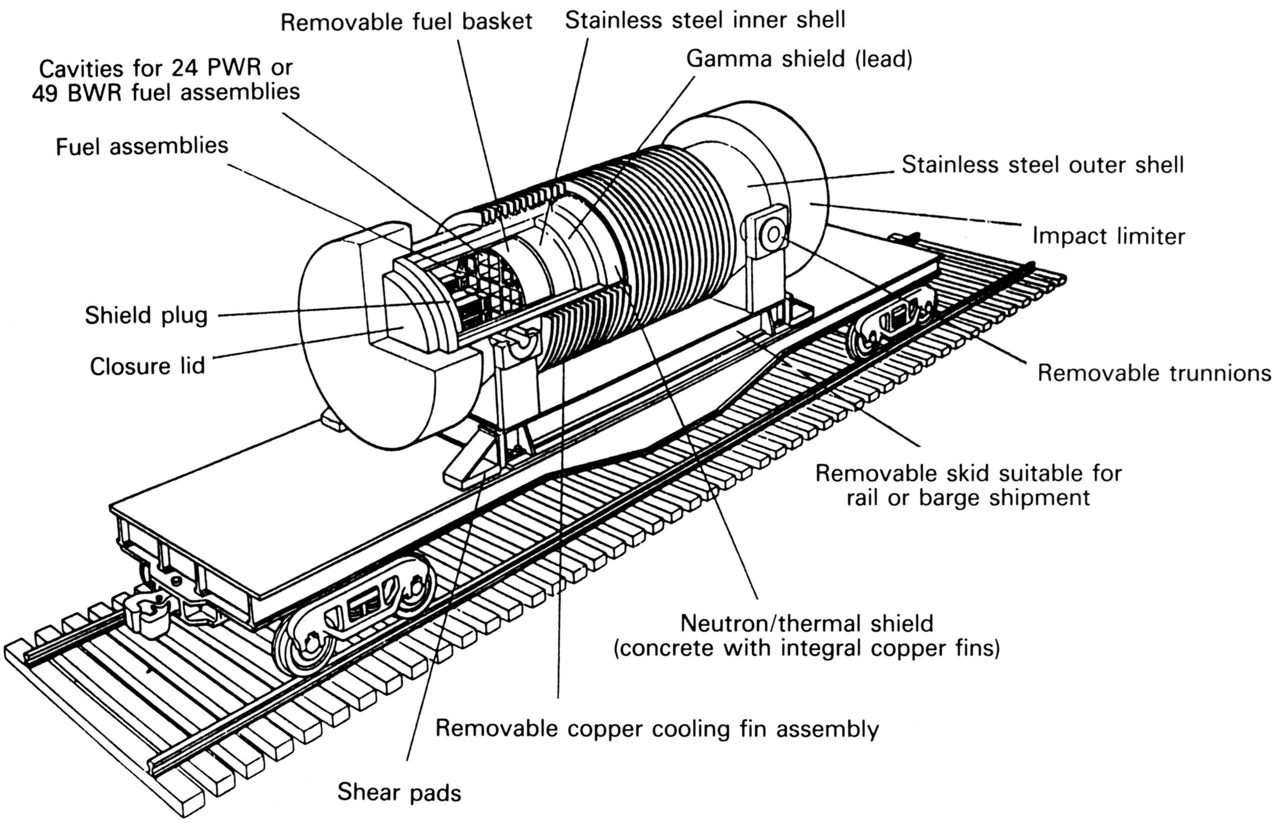


Fig. 19. The BR-100 rail/barge transportation cask (designed by B&W).

of full-scale tests involving new cask designs. These tests, based on the standards shown in Fig. 20 (Ref. 68), were conducted to verify computer models of cask damage in postulated accidents. The following are examples of the tests:

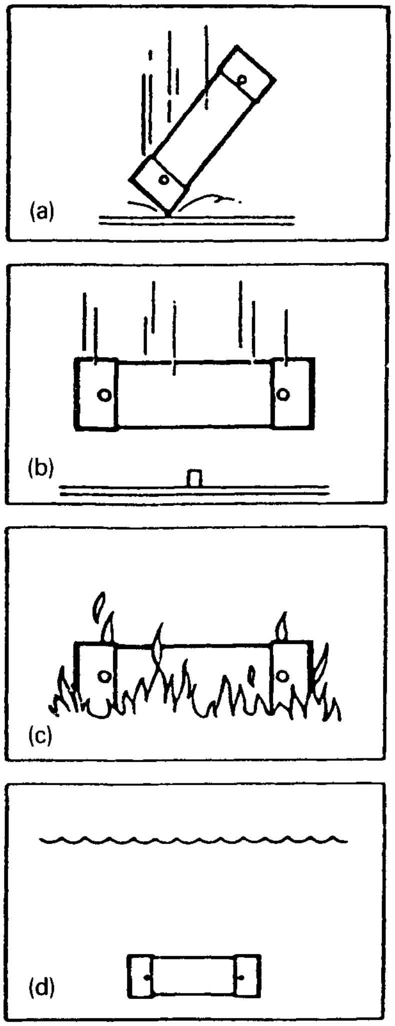
1. *Crash tests:* A tractor trailer truck carrying a cask was crashed into a massive concrete wall at 61 and 84 mile/h. No damage was observed on the cask in either crash. However, the cask was deformed as predicted by the calculations. No radioactivity would have been released. There was leakage of nonradioactive coolant fluid, but not until the cask was lifted from the wreckage.

2. *Locomotive test:* A locomotive traveling at 80 mile/h rammed broadside into a cask mounted on a truck. The locomotive was severely damaged but the cask was only slightly dented, almost exactly as predicted by the computer simulation of the test. No radioactivity escaped to the environment.

3. *Crash fire test:* A railroad car carrying a cask was crashed into a massive concrete barrier at 81 mile/h. Then the cask and the railroad car were subjected to an intense jet fuel fire for 125 min, which totally engulfed them. As a result of the fire, the lead shield between the inner and outer walls melted. After 100 min, the pressure from the molten lead eventually caused a small crack (~0.004 in. wide) through which some molten lead escaped. No radioactivity escaped. The cask design was corrected after this test to prevent the escape of molten lead.

4. *Drop test:* A cask formerly used for shipping spent fuel from a research reactor was dropped from a helicopter and crashed on the desert ground at 235 mile/h. Although the cask was buried in >4 ft of the hard-packed soil, its only damage consisted of paint scratches.

Scientists at SNL also simulated terrorist attacks using explosive devices. The detonation of an explosive on a full-scale spent-fuel shipping cask indicated that <1% of the contents would be released to the environment. The weight of the cask should discourage would-be hijackers from attempting to steal it by removing it from the carrying vehicle. Removal of the fuel itself from the cask would likewise be difficult, since the closure plug weighs several tons. If the hijackers succeded in removing the plug, they could receive a lethal radiation dose from the exposed fuel assemblies.



For certification by the NRC, a cask must be able to withstand a series of accident conditions. These conditions were developed in a National Academy of Sciences committee's recommendations on tests that would simulate damage to spent-fuel casks in the most severe credible accidents. The mechanical tests (free drop and puncture), the thermal (fire) test, and the water immersion (3-ft) test are performed in sequence to determine the cumulative effects on one package. A separate cask is subjected to the deep water immersion (50-ft) test. Paraphrased descriptions of the regulatory tests follow.

Mechanical

a. Free drop—30-ft drop of the spent-fuel cask onto a flat, horizontal, unyielding surface* with the cask positioned so that its weakest point is struck.
b. Puncture—40-in. free drop of the cask onto a 6-in.-diam steel bar at least 8 in. long; the bar is to strike the cask at its most vulnerable spot.

Thermal

c. Fire—After the mechanical tests are completed, the package is totally engulfed in a fire or furnace at 1475°F for 30 min.

Water immersion

d. Immersion of all packaging surfaces under at least 3 ft of water for 8 h; immersion of entire packaging under 50 ft of water for 8 h.

*The "unyielding surface" criteria require that the result of the impact be borne completely by the cask. Drops from heights 2.5 to 3 times greater onto normal hard surfaces would be comparable. Drop tests have been conducted at much greater heights (2000 ft) without breaching the cask containment.

Fig. 20. Standards spent-fuel transportation casks must satisfy.

XIV. THE LOW-LEVEL RADIOACTIVE WASTE POLICY ACT AND ITS 1985 AMENDMENTS

All 50 states generate LLW regardless of whether or not they have operating nuclear power or research reactors within their borders. A large fraction of LLW is generated by research establishments, industry, and hospitals that use radioisotopes. When the Low-Level Radioactive Waste Policy Act[69] (LLRWPA) and its amendments[70] were passed in 1980, there were only three sites open to accept LLW, and the same sites are the only ones available today in 1990. They are located in Barnwell, South Carolina; Beatty, Nevada; and Richland, Washington. The governors of those states pressed Congress to do something about LLW disposal; they threatened to reject LLW coming from anywhere outside their borders. Congress responded by passing the LLRWPA in 1980.

The LLRWPA dictated that each state is responsible for the disposal of all the LLW generated within its borders. The act assumes that the LLW can be safely and efficiently managed and stored on a regional basis. To carry out the disposal of LLW, the states may enter into "compacts" with other states to establish and operate regional disposal facilities, subject to congressional approval. A compact will be a legal entity created on the basis of a contract signed by all the member states. It will be acting through a "compact commission" created by the state for that purpose. The authority of the compact commission will depend on whether or not the host state (where the site is located) is an agreement state.

If the host state is an agreement state, then it can license and regulate its LLW site. If not, the NRC will license the facility. The licensing requirements for LLW sites are given in 10CFR61 (Ref. 71).

No compact can take effect without congressional consent. Congress may, by law, withdraw its consent every 5 yr after the compact goes into effect. After January 1, 1986, any such compact could have restricted acceptance of LLW from within its region only and allow rejection of requests for disposal of any LLW generated elsewhere.

Immediately after the passage of the LLRWPA, the states began the process prescribed by the act in an attempt to comply with it, but it became clear well before the deadline of January 1, 1986, that no compact would be ready for approval by that date. For this reason, Congress initiated and passed amendments to the act in December 1985, and the President signed them into law on January 15, 1986.

The 1985 amendments modified the LLRWPA in two important aspects. First, a transitional period was established from January 1, 1986, to December 31, 1992, during which time limited access to the existing disposal sites would be provided, until compacts are approved and start operating. Second, and more important, the amendments prescribe penalties in the form of surcharges for states that will not meet the specified milestones. The surcharges range from \$10.00 to 120.00/ft^3 LLW. The DOE will hold 25% of the surcharge payments in an interest-bearing account. The various state compacts will receive rebates from that account, depending on the progress made toward selecting, preparing, and licensing a site.

A national limit of 2.8 million ft^2 LLW/yr was established for disposal at the existing facilities. In order not to exceed the limit, the LLRWPA imposes an annual limit on LLW generated by the electric utility industry and encourages nonutility generators to reduce the volume of their wastes.

The main milestones specified by the LLRWPA amendments are as follows:

1. By July 1, 1986, a compact should be ratified or intent to develop a site should be certified. Failure to meet this deadline will result in doubling of the surcharge from \$10 to 20/ft^3 for 6 months. All surcharge rebates will be lost. Access may be denied after January 1, 1987.

2. By January 1, 1988, the host state should be identified, a developer should be selected, and a siting plan should be prepared. Failure to meet this deadline will result in doubling the surcharge for 6 months, and quadrupling it for the next 6 months. All surcharge rebates from July 1, 1986, to January 1, 1988, will be lost. Access to LLW sites may be denied after January 1, 1989.

3. By January 1, 1990, an application for an operating license for a disposal site must be submitted, by an individual state or through a compact, or the governor should certify in writing to the NRC that the state will be able to provide for proper management and disposal of its LLW. If this milestone is not met, all surcharge rebates between January 1, 1988, and January 1, 1990, will be lost, and access to disposal facilities may be denied after January 1, 1990.

4. By January 1, 1992, the governor's certification will not be enough. A license application must be submitted, for a state or a compact to be in compliance. The consequences of noncompliance are a tripling of the surcharge rate to \$120/ft, until a license application is submitted, and denial to a LLW site after January 1, 1993, by licensed compacts.

5. By January 1, 1996, all states must have a LLW disposal site. No surcharge rebate payments will be made after this date. Each state will be fully responsible for all LLW generated within its borders.

The compact groupings at this time (1990) appear as shown in Fig. 21. All states, except Vermont and Puerto Rico, have met the January 1, 1990, deadline for compliance with the LLRWPA amendments. The next major deadline is January 1, 1992, when a license application for a LLW facility must be submitted to the NRC.

One provision of the LLRWPA amendments is a directive to the NRC to "establish standards and procedures . . . to exempt specific radioactive waste streams from regulation by the Commission due to the presence of radionuclides in such streams in sufficiently low concentrations or quantities as to be below regulatory concern." Pursuant to this directive, the NRC is in the process of defining LLW that may be characterized as "below regulatory concern" (BRC) waste. In 1989, the term BRC waste was replaced by "exemption from regulatory control."

XV. DISPOSAL OF LLW

Low-level waste can be subdivided into three physical forms: liquids, wet solids, and dry solids. Radionuclides present in LLW differ depending on the source of the wastes. Nuclear reactor operations generate LLW containing activation products and a small quantity of fission products and, in rare cases, TRU isotopes. Industrial and institutional (hospitals and research laboratories) LLW contain ^{3}H, ^{14}C, ^{60}Co, ^{137}Cs, ^{99}Tc, ^{22}Na, and others.

Liquid LLWs are fluids that have been contaminated with radioactive materials. Examples are chemical regenerative solutions, decontamination solutions, liquid scintillators, contaminated oils, and other miscellaneous organic and inorganic fluids. The liquids are produced as a result of several different operations at a nuclear power plant. Examples are ion-exchange resins; contaminated oils from oil-water separators, reactor coolant pumps, or hydraulic scrubbers; liquid scintillators used for radiation monitoring; laundry waste streams; and discharges from equipment and floor drains.

Wet solids are relatively thick slurries containing a certain fraction of solids. Examples are evaporator bottoms, spent ion-exchange resins, expended loaded

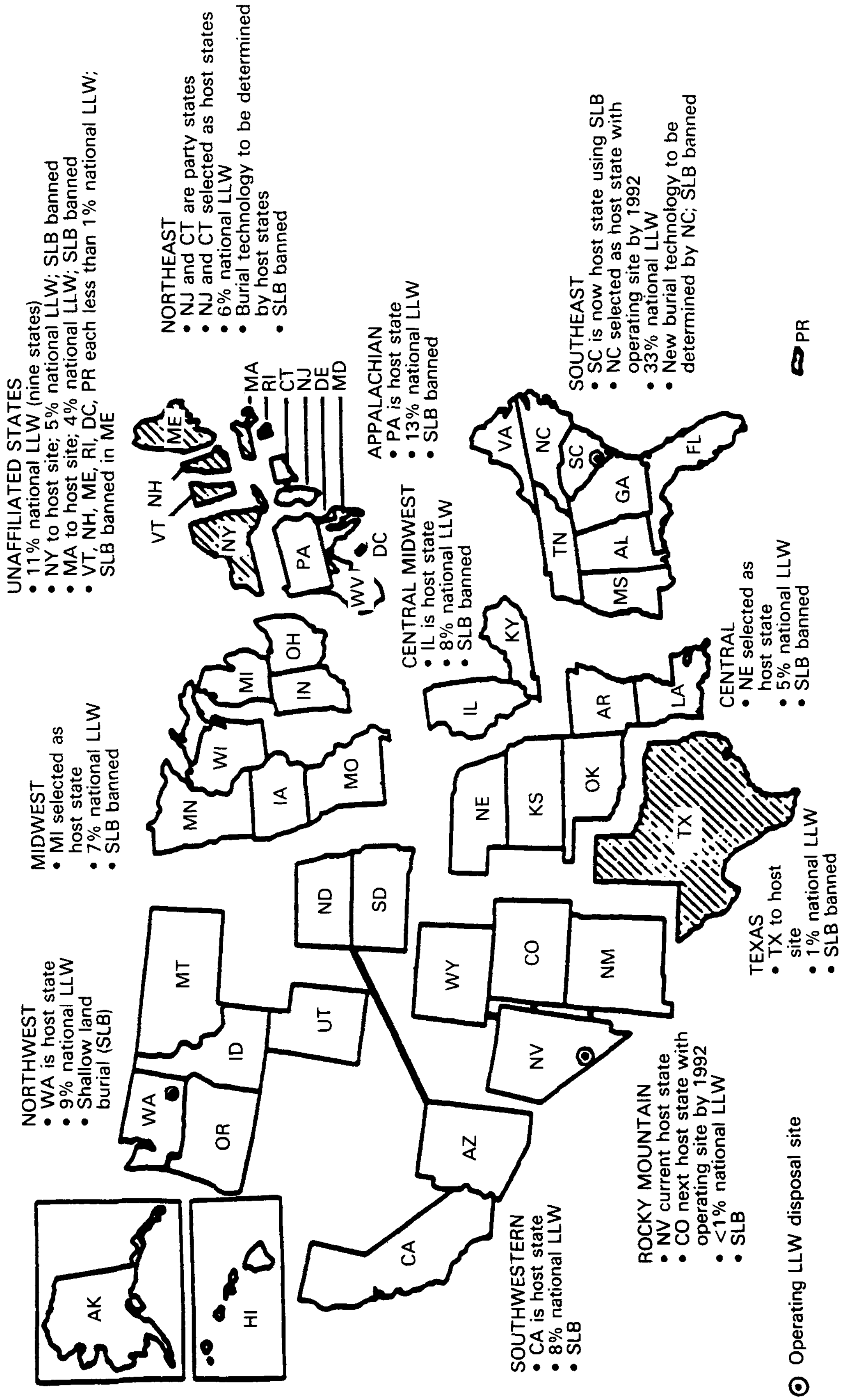


Fig. 21. Map of the U.S. LLW compacts as of 1990.

filter cartridges, and others. From research programs and medical establishments, one gets animal carcasses, tissues, excreta, biological cultures, etc.

Dry solids are trash; contaminated clothing, tools, and equipment; and irradiated hardware.

Radioactive waste packages that may be buried in near-surface disposal sites are divided into three classes—A, B, and C—according to the type, activity concentration, and the half-lives of the nuclides contained in the package.[71] In general terms, class A is the waste with the least radioactivity; it becomes nonhazardous during the institutional control period after the site is closed. Institutional control means that the site is controlled by a state or federal agency. Class B waste is more radioactive than class A and remains hazardous for up to 300 yr. Class C is the waste with the maximum allowed radioactivity for near-surface burial; it remains hazardous for >300 yr and it may require burial at greater depths than class A or B.

The classification procedure will be better understood with the help of Tables XV.a and XV.b, reproduced from 10CFR61 (Ref. 71). Consider a package that contains only radioisotopes that are listed in Table XV.a. The waste is class A if the concentration is <0.1 times the values given in Table XV.a. The waste is class C if the concentration exceeds 0.1, but is less than the values given in Table XV.a.

If the waste package does not contain any of the nuclides listed in Table XV.a, the classification is determined with the help of the concentrations given in Table XV.b. The waste is class A if the concentration does not exceed the value given in column 1. The waste is class B if the concentration exceeds the value in column 1 but does not exceed the value in column 2. The waste is class C if the concentration exceeds the value in column 2 but does not exceed the value in column 3.

The waste is class A if it does not contain any of the nuclides listed in Tables XV.a or XV.b. Packages containing the isotopes indicated by an asterisk in columns 2 and 3 of Table XV.b will be class B no matter what their concentrations are, unless other isotopes dictate a different classification. The concentrations of these isotopes are limited by effects such as external radiation and internal heat generation levels, dictated by transportation, handling, and disposal requirements.

If the waste exceeds the concentrations given in Table XV.a or in column 3 of Table XV.b, disposal at near-surface sites is not permitted. Such waste is called "greater than class C" (GTCC): Under the LLRWPA amendments, GTCC waste will be the responsibility of the federal government. The NRC has amended its regulations to require disposal of GTCC waste in a deep geologic repository unless the NRC approves an alternate disposal. This rule became effective in June 26, 1989. With this new rule in effect, the DOE will have to place such wastes in the HLW repository or build a special one for "intermediate-level wastes."

If a package contains a mixture of radionuclides, the sum of the fractions should be used to determine the class. The sum of the fractions of the activities of individual isotopes from a particular column must be <1 if the waste is to be classified based on that column. For mixtures of radionuclides from both tables, the most restrictive classification applies.

The disposal requirements for class A, B, or C wastes are described in 10CFR61, specifically in paragraph 61.56. The major requirements are the following:

1. The waste must not be packaged for disposal in cardboard or fiberboard boxes.

2. Liquid waste must be solidified or packaged in sufficient absorbent material to absorb twice the volume of the liquid.

TABLE XV.a

Activity Concentrations Used for the Classification of LLW

Nuclide	Concentration
^{14}C	8 Ci/m^3
^{14}C in activated metal	80 Ci/m^3
^{59}Ni in activated metal	220 Ci/m^3
^{94}Nb in activated metal	0.2 Ci/m^3
^{99}Tc	3 Ci/m^3
^{129}I	0.08 Ci/m^3
Alpha-emitting TRU with $T > 5$ yr	100 nCi/g
^{241}Pu	3 500 nCi/g
^{242}Cm	20 000 nCi/g

TABLE XV.b

Activity Concentrations Used for the Classification of LLW

Nuclide	Concentration (Ci/m^3)		
	Column 1	Column 2	Column 3
All nuclides with $T < 5$ yr	700	*	*
^{3}H	40	*	*
^{60}Co	700	*	*
^{63}Ni	3.5	70	700
^{63}Ni in activated metal	3.5	700	7 000
^{90}Sr	0.04	150	7 000
^{137}Cs	1	44	4 600

*No limits established for these nuclides in columns B or C. Other considerations (external radiation or internal heat generation) may limit their concentrations. These wastes will be class B unless other considerations determine them to be class C.

3. Liquid contained in solid waste shall not exceed 1% of the waste volume.

4. Waste must not be explosive or reactive at normal temperatures and pressures.

5. The waste must not contain or be capable of generating toxic gases, vapors, or fumes harmful to people. (This does not apply to gaseous waste; see item 7 below.)

6. Waste must not be pyrophoric.

7. Gaseous waste must be packaged at a pressure <1.5 atm at 20°C and total activity must be $<3.7 \times 10^{12}$ Bq (100 Ci)/container.

8. Wastes containing hazardous, biological, pathogenic, or infectious material need special treatment.

To reduce storage space and, more important, to reduce disposal charges, calculated in dollars per cubic foot, volume reduction of LLW appears to be an attractive procedure. All generators of LLW apply volume reduction to all the wastes that can be treated in this manner. One should be careful, however, because the volume reduction may shift the waste from one class to a higher one.

Volume reduction is achieved in one of three ways: compaction, incineration, or evaporation. Studies have been made[72,73] on the best ways to reduce the volume, and many commercial firms offer their services or sell the machines that achieve this goal. Companies that offer commercially available compactors advertise compaction ratios of more than a factor of 10. The bulk density of solid LLW is, on the average, ~160 to 240 kg/m^3 (10 to 15 lb/ft^3). The compacted one may have a density of >1000 kg/m^3.

Incineration, used when the equipment is available and the method is allowed for the wastes at hand, offers even greater volume reduction than compaction. The ashes of the process offer a chemical composition that immobilizes the radioactivity. Usually, the ashes are compacted into a solid before disposal.

Evaporation is another method offered commercially for the concentration of radioactive liquid effluents. Finally, there are commercial radioactive waste volume reduction systems available that combine evaporation, incineration, and compaction.

Certain LLW can be solidified. Examples are ion-exchange resins, filter sludge, chemical concentrates, decontamination solutions, contaminated oil, and fuel fabrication sludges. The solidification media are cement, bitumen, urea formaldehyde (no longer permitted in the United States), and Dow media (a polyester-styrene type material provided only by the Dow Chemical Company).

The advantages of the stabilizing medium vary in each case. For example, the use of bitumen employs an inexpensive material, offers high volume reduction because the water is evaporated first, and produces blocks of waste with excellent nonleachability and ageing qualities. The use of cement (silicate or portland) produces a solid with nonleaching characteristics similar to or better than those of plain portland cement. By introducing ion-exchange additives, the leaching properties can be further improved.

The market offers several systems. The specific system to be installed depends on the type and volume of wastes and the type of generator (nuclear reactor, hospital, or research establishment).

After the LLW is treated and packaged, it is shipped to the disposal site. As mentioned earlier, there are at present (1990), only three LLW disposal sites in the United States. In all three, the wastes are placed in open slit trenches (Fig. 22) ~300 m (1000 ft) long, 30 m (100 ft) wide, and 6 to 15 m (20 to 50 ft) deep, sloped toward the open end. A few feet of sand or gravel are placed at the bottom of the trench before the wastes are put in place. For sites designated as "shallow sand burial banned," the wastes must be placed in underground concrete bunkers: The trench alone is not enough to guarantee isolation; a man-made barrier is also required. Filled trenches are backfilled with sand, capped with clay, and finally covered with topsoil on which vegetation is grown. The trenches are marked with permanent monuments that describe the contents, the boundaries, the dates during which the trench was open, and other pertinent information.

Open-slit trenches are not the only type of LLW disposal site. In France, for example, LLW is disposed on the surface, in tumuli covered with clay and earth (Fig. 23), or in trenches in concrete monoliths (Fig. 24). The choice of method depends on the type and activity level. The LLW is placed in tumuli if the type and contents of the package can guarantee safe disposal by this method. Most reactor-produced LLW belongs to this category. This waste is delivered in metal drums or concrete containers. If the packaging does not provide adequate protection against the radioactivity, the waste is placed first in a concrete monolith that provides the extra protection needed. Wastes treated in this way are delivered in metal drums or in boxes. Certain "advanced" technologies, having mostly to do with new methods of packing, are under consideration and may be used in the new sites to be developed by the state compacts.

All countries with nuclear activities of any kind have a program for the disposal of LLW, with one option being shipment to another country, by agreement. In general, the big LLW producers have instituted programs similar to those followed in the United States, namely, burial of the wastes in shallow ground (France), or at a depth of 30 to 60 m (Sweden and Finland) (Ref. 74), or in unused mines (United Kingdom). In Japan, a LLW facility with a design capacity of

Fig. 22. Open-trench LLW site (Richland, Washington).

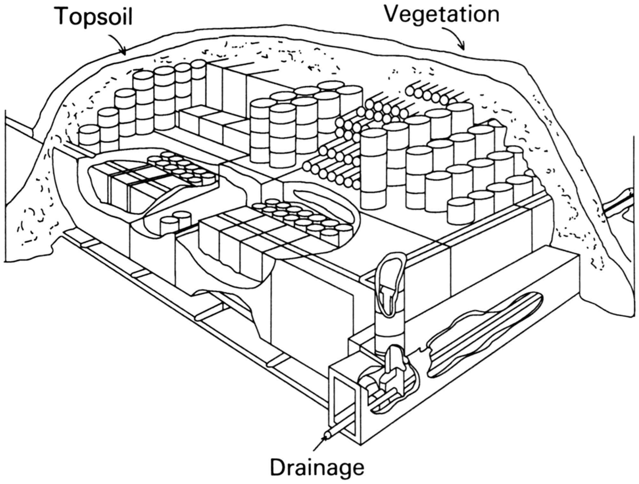


Fig. 23. French tumulus concept used for the disposal of LLW.

600 000 m^3 will be ready for operation in 1991. In general, the Japanese approach will be similar to that of France and the United States, namely, shallow disposal for LLW and deep geologic disposal for HLW.

XVI. DECOMMISSIONING OF NUCLEAR POWER PLANTS

Decommissioning is the term used to encompass all the activities necessary to take place at the end of life (EOL) of a plant so that the site where the plant is located does not pose a hazard to the public and can, therefore, be considered as a property for unrestricted use [see 10CFR50 (Ref. 75)]. Decommissioning is required not only for nuclear power plants, but for every industrial facility using or producing radioactive materials that reaches the end of its useful life. The decommissioning of a nuclear facility is different from that of other industrial plants because of the radioactivity involved.

The nuclear facilities mentioned by 10CFR50 include nuclear reactors and chemical plants handling radioactive materials such as enrichment plants, fuel fabrication plants, and reprocessing plants. Of the two types of nuclear facilities, it is reactor decommissioning that is the topic of this section. The decommissioning of chemical-type facilities essentially amounts to decontamination.

A nuclear power plant is licensed for 30 yr. Although there is activity in progress to extend the life beyond the original licensing period, there is also the

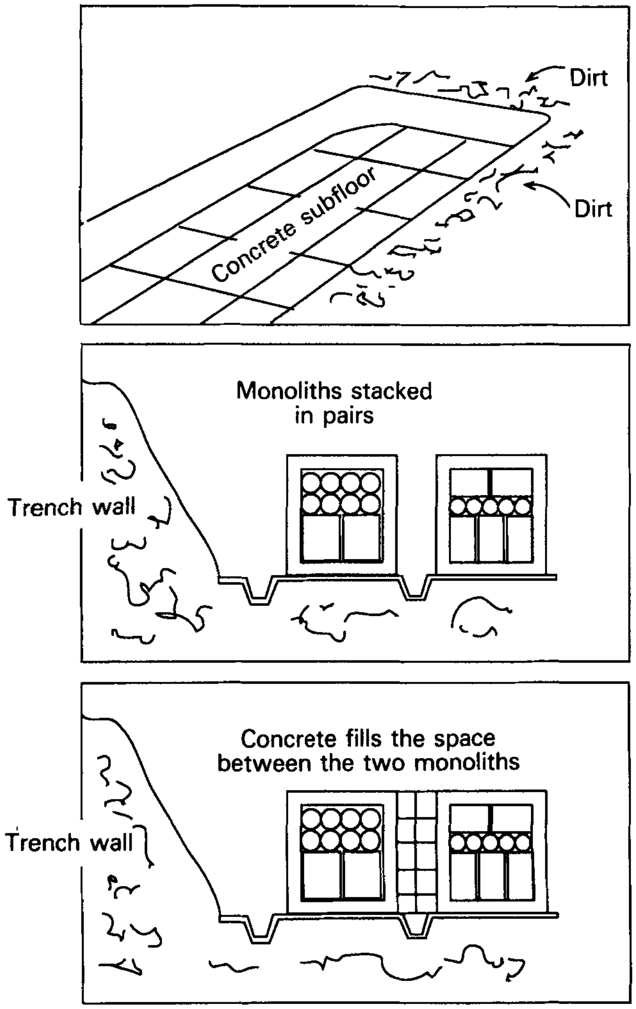


Fig. 24. Disposal of French LLW using concrete monoliths.

possibility of shutting down a plant prematurely for various reasons, such as inefficient operation, expensive repairs, or expensive decontamination due to an accident. In any event, nuclear plants have a finite lifetime and every utility that owns a commercial nuclear power plant is obligated by law to make plans for decommissioning. A new rule published by the NRC in June 1988, in 10CFR50, requires that every holder of an operating license "on or before July 26, 1990 . . . shall submit information in the form of a report, . . . indicating how reasonable assurance will be provided that funds will be available to decommission the facility."

A nuclear utility is, therefore, required to prove that when the reactor is shut down for the last time, a well planned, organized effort will commence that will remove hazardous materials from the site, and eventually, unless there are other reactors on the same site, it will return the site to unrestricted use. To help the utilities achieve this goal, the NRC addresses decommissioning in many parts of the Code of Federal Regulations in addition to 10CFR50, specifically in 10CFR parts 20, 30, 40, 51, 70, and 72. The NRC also supported various decommissioning studies[76,77] as did the Atomic Industrial Forum[78] and the Electric Power Research Institute.[79,80]

The disposal of spent fuel is not considered part of decommissioning. When a plant is shut down at the EOL, one of the first tasks is the shipment of all the spent fuel to a federal facility (either the MRS or the repository) to be disposed of in accordance with the NWPA. With the spent fuel out of the picture, most of the radioactive materials involved in decommissioning are LLW, with possibly a small amount of TRU waste in rare cases where fuel assemblies leaked and some TRU elements escaped from the fuel rods. Note that ∼85% of the volume of a nuclear power plant never becomes radioactive.

There are three decommissioning methods considered at this time, known as prompt dismantling, mothballing, and entombment. In the United States, they are denoted by the acronyms DECON, SAFSTOR, and ENTOMB, respectively. A brief description of each method follows:

1. *DECON:* All the radioactive materials, such as components and structures, are removed relatively soon after final shutdown. Upon completion of the operation, the nuclear license is terminated and the site is released for unrestricted use.

2. *SAFSTOR:* All liquid radioactive materials and all portable solid radioactive materials are removed. The remaining structures and equipment are secured and continuously monitored to ensure protection of the public. Use of the facility is controlled by an amended nuclear license. Eventually, the rest of the plant will be dismantled and the license will be terminated as with DECON.

3. *ENTOMB:* This is the same as SAFSTOR except that the facility is sealed up completely, usually by covering everything with concrete, thus creating one monolithic structure. No accessible area is contaminated and the entombed radioactive materials are monitored for as long as necessary to permit the activity to decay to such a level that the site can be returned to unrestricted use.

The method to be used depends on many factors, but the two most important ones are cost and the total radiation dose to workers and the public. The method depends on the utility plans for the site. If the site has only one reactor and the utility has no plans to build others, DECON appears to be the best method. If, on the other hand, the site has more than one reactor or the utility plans to build additional power plants, SAFSTOR may be attractive since the site will not be released for unrestricted use anyway, and monitoring by the personnel who work on the site will not incur much additional expense. SAFSTOR will reduce both occupational doses and amounts of radioactive wastes to be shipped for disposal, relative to DECON. Of the three methods, ENTOMB seems to be the least attractive because the licensee must demonstrate that the site could be released for unrestricted use after ~100 yr, following shutdown, thus requiring a long-term commitment.

There are two general types of radioactive materials involved in decommissioning: contamination carried by the water coolant circulating through the core and activation products produced by neutron absorption both inside and outside the pressure vessel.

The major activation products are in the pressure vessel and the reactor internals. At shutdown, there are many isotopes present, most of them short-lived (Table XVI). From the half-lives involved, it is obvious that the most important radioisotopes are ^{60}Co, ^{59}Ni, and ^{94}Nb, and in the case of the high temperature gas cooled reactor, one should add ^{14}C. Cobalt-60 emits the most penetrating radiation but has the shortest half-life. The radiation doses to which radiation workers and members of the public will be exposed are estimated to be[81] 9 to 19 person-Sv (900 to 1800 man-rem). Almost all of this dose will be received by radiation workers; the dose to the public is <0.07 person-Sv (7 man-rem).

Major radioactive components that are treated as LLW are as follows[82]: heat exchangers whose inside surfaces are contaminated and will probably be shipped intact to the LLW disposal site; instrumentation cables that will be coiled and shipped in containers to the LLW site; steam generators that will probably be shipped intact after their openings are welded shut; reactor coolant pumps whose inside surfaces are contaminated will be shipped intact; the pressurizer, which, after the interior has been cleaned, will be shipped to the LLW site, serving as its own container; the biological shield, which will be taken apart before shipping; and the reactor pressure vessel, which is the largest component and contains the largest amount of radioactivity. The total volume of materials involved (activated matter, contaminated equipment, and decontaminating materials) is estimated to be, depending on the decommissioning method, from 13 500 to 16 000 m^3 (475 000 to 560 000 ft^3).

TABLE XVI

Most Important Radioisotopes Encountered During Decommissioning

Isotope	Half-Life (yr)	Radiations Emitted[a]
^{60}Co	5.27	1.17- and 1.33-MeV gamma 0.314- and 1.48-MeV beta
^{90}Sr	80	
^{63}Ni	100	
^{14}C	5 740	0.158-MeV beta
^{94}Nb	20 000	0.702- and 0.871-MeV gamma 0.490-MeV beta
^{59}Ni	80 000	Bremsstrahlung up to 1.06 MeV and cobalt X rays

[a] Radiations are listed only for the isotopes of concern.

If further decontamination of components or structures is deemed necessary after common janitorial practices such as vacuuming, sweeping, mopping, and scrubbing with ordinary detergents are used to remove as much of the radioactivity as possible, it will be carried out by one of the following methods or processes: chemical, physical, electropolishing, and ultrasonic.

The financing of decommissioning is regulated by the federal government with the main objective being the assurance that whatever plan a utility adopts, that plan will be workable and, most important of all, sufficient funds will be available to complete the decommissioning operation, no matter when the plant is shut down. That is, the financing of decommissioning should be such that it can be carried out even if the plant is shut down before its planned EOL.

Decommissioning financing methods fall into three general types:

1. The licensee deposits the necessary funds, either at start-up or during the plant life, into an account separate from the licensee's assets and outside the licensee's control. The fund so accumulated is called an external fund.

2. The licensee holds the funds as part of the company's assets and under its control. This fund is called an internal reserve.

3. The licensee provides some form of guarantee that the funds will be available (e.g., insurance).

From the point of view of the NRC, the preferred funding is that which provides the total amount needed for decommissioning at the beginning of the life of the plant and is not under the control of the licensee (method 1). Then, no matter what happens to the licensee's assets and regardless of when the plant will be shut down (after 30 yr or any time earlier), decommissioning is certain to be completed satisfactorily.

The proposed NRC rules also require that the licensee shall provide detailed decommissioning plans for the plant within 2 yr after operations cease or 1 yr prior to the expiration of the license, whichever comes first. The plans must contain sufficient detail to show that decommissioning can be accomplished safely. In particular, the plans should address the disposal of the radioactive wastes (i.e., existence of a disposal site) and the existence of adequate funds. The licensee will be required to maintain detailed records on all aspects of decommissioning and will be assigned a time period within which the task should be accomplished. All activities must obey the as-low-as-reasonably achievable (ALARA) principle.

The cost of decommissioning a large commercial nuclear power plant is estimated by the NRC to be in the range of $100 to 140 million in 1986 dollars. The nuclear industry[83] gives an estimate of $180 to 210 million in 1988 dollars. Table XVII gives the separate estimates for the three methods considered. More details concerning decommissioning costs can be found in Refs. 76, 77, 79, and 83.

Decommissioning experience exists in the United States as well as in other countries as a result of dismantling many research and small experimental power reactors during the last 25 yr. Table XVIII shows reactors dismantled in the United States. One could add to this list the reactors decommissioned or being decommissioned in the United Kingdom, France, Germany, Canada, and Japan.

TABLE XVII

Estimated Decommissioning Costs for a PWR*

(Millions of 1988 dollars)

	DECON	SAFSTOR (30 yr)	ENTOMB (100 yr)
Preliminary activity		30	80
Surveillance		35	10
Removal	180	160	120
Total	180	225	210

*Taken from Ref. 82.

In the United States, the first decommissioning of a commercial nuclear power plant was that of Shippingport. That project was carried out by the DOE using the DECON method. The work started in January 1985 and was completed in 1989 (Ref. 84). The estimated cost of decommissioning the Shippingport reactor was $98 million in 1986 dollars.

Is there technology available for safe decommissioning of nuclear power plants? The nuclear industry and federal government certainly think so. Reactors that have been decommissioned were small in size, but the increase in size does not require the introduction of new technology. If any new technology is introduced, it will be in the area of robotics, which will allow remote handling of most activities and thus reduce accumulated radiation dose to personnel. There is controversy about the cost. The critics of the nuclear industry estimate it to be about $1 billion, which is about five times more than the industry's estimate.

XVII. TRANSPORTATION OF RADIOACTIVE MATERIALS

The regulations pertaining to transportation of radioactive materials are issued by three federal agencies: the Department of Transportation (DOT), the NRC, and the U.S. Postal Service. The DOT rules are found in 49CFR100 through 177 and 49CFR178 through 199 (Ref. 85). The NRC regulations are found in 10CFR171. Finally, the postal service regulations are found in 39CFR124.

TABLE XVIII

Decommissioned Reactors

Plant	Location	MW(electric)	Start-Up	Shutdown	Type of Decommissioning
SRE	Santa Susana, California	6	1957	1964	Dismantle
Vallecitos	Pleasanton, California	5	1957	1963	Mothball
Shippingport	Shippingport, Pennsylvania	72	1957	1982	Dismantle
Dresden 1	Morris, Illinois	200	1959	1984	Mothball
Saxton	Saxton, Pennsylvania	7	1962	1972	Mothball
Hallam	Hallam, Nebraska	75	1962	1964	Entomb
Elk River	Elk River, Minnesota	22	1962	1968	Dismantle
Humboldt Bay 3	Eureka, California	65	1963	1983	Mothball
CVTR	Parr, South Carolina	17	1963	1967	Mothball
Piqua	Piqua, Ohio	11	1963	1966	Entomb
Enrico Fermi 1	Lagoona Beach, Michigan	61	1963	1973	Mothball
Pathfinder	Sioux Falls, South Dakota	59	1964	1967	Mothball
BONUS	Punta Higuera, Puerto Rico	17	1964	1968	Entomb
Peach Bottom 1	Peach Bottom, Pennsylvania	40	1966	1974	Mothball
LaCrosse	Genoa, Wisconsin	50	1969	1987	Mothball

Federal rules and regulations change frequently. Persons routinely involved with transportation of radioactive materials should be certain that they have and apply the current ones. This section is based on the 1983 revisions of the DOT and the 1984 revisions of the NRC regulations.[52,85]

There are also international bodies that deal with the transportation of radioactive materials, the primary one being the International Atomic Energy Agency (IAEA). The IAEA has been the primary agency for the establishment of regulations governing the transportation of such materials, regulations that formed the basis of most international agreements on the subject.

In the United States, there is >30 yr of experience in the transportation of radioactive materials, both HLW and LLW. As a result of careful selection of the packages used to transport the materials and the special precautions taken by the carrier both in terms of the vehicle and the route followed, no member of the public has been injured by release of radioactivity in an accident involving radioactive cargo.

Definitions and classifications, regarding the transportation of radioactive materials, are given in the May 1984 revision of 10CFR71. An excellent presentation of the topic is given in Ref. 63. Given below is a selection of those terms that are the most frequently encountered. For more details the reader is directed to consult 10CFR71.

1. *Source material* is uranium or thorium or any combination of the two in any form, that contains by weight $\frac{1}{20}$ of 1% (0.05%) or more of uranium, thorium, or any combination of the two.

2. *Special nuclear material (SNM)* is plutonium; ^{233}U; uranium enriched to the isotope 233 or 235; any other material that the NRC determines to be SNM, but does not include source material; or any other artificially enriched material.

3. *By-product material* is material that became radioactive after being exposed to radiation generated in the process of utilizing SNM. Note that materials becoming radioactive by irradiation in an accelerator are not considered by-product material.

4. *Special form radioactive material* is radioactive material that satisfies the following conditions:

   a. It is either a single solid piece or is contained in a sealed capsule that can be opened only by destroying the capsule.

   b. The piece or capsule has at least one dimension not less than 5 mm.

   c. It satisfies the test requirements of paragraph 10CFR71.75 (this paragraph specifies certain tests the package must satisfy, e.g., drop tests, bending, heat, etc.).

An example of a special form material is a solid metal source sealed in a high-integrity container.

5. *Normal form radioactive material* is material that does not qualify as special form radioactive material. Examples are waste material in a plastic bag, liquid in a bottle within a metal container, powder in a glass or plastic bottle, and gas in a cylinder.

6. *Low-specific-activity (LSA) material* is any of the following:

   a. uranium or thorium ores and physical or chemical concentrates of those ores

   b. unirradiated natural or depleted uranium or unirradiated natural thorium

   c. tritium oxide in aqueous solution, provided the concentration does not exceed 1.85 × 10^8 Bq/ml (5.0 mCi/ml)

   d. material in which the radioactivity is essentially uniformly distributed and in which the estimated average concentration per gram of contents does not exceed the following limits, given with reference to activities specified as A1 and A2 in the Appendix of 10CFR71 and reproduced here for selective isotopes in Table XIX.

      i. 3700 Bq (0.0001 mCi) from radionuclides with A2 activity not >1.85 × 10^9 Bq (0.05 Ci)

TABLE XIX

Values of A1 and A2 Activities for Selected Nuclides*

Isotope	A1 Special Form (Ci)	A2 Normal Form (Ci)
^{3}H (in H_2O form)	1000	1000
^{14}C	1000	60
^{137}Cs	30	10
^{60}Co	7	7
^{99}Mo	100	20
^{123}I	50	50
^{129}I	1000	2
^{201}Pb	20	20
^{210}Pb	100	0.2
^{226}Ra	10	0.05
^{233}U	100	0.1
^{235}U	100	0.2
^{239}Pu	2	0.002
^{252}Cf	2	0.009

*Taken from Ref. 52.

ii. 1.85×10^5 Bq (0.005 mCi) from radionuclides with A2 activity $>1.85 \times 10^9$ Bq (0.05 Ci) but $<3.7 \times 10^{10}$ Bq (1 Ci)

iii. 1.11×10^7 Bq (0.3 mCi) from radionuclides with A2 activity $>3.7 \times 10^{10}$ Bq (1 Ci)

e. nonradioactive objects externally contaminated with radioactive material, provided that the radioactive material is not readily dispersible and the surface contamination, when averaged over a 1-m^2 area, does not exceed 3.7×10^7 Bq/m^3 (0.0001 mCi/cm^3) for which the A2 quantity is $<1.85 \times 10^9$ Bq (0.05 Ci), or 3.7×10^8 Bq/m^3 (0.001 mCi/cm^3) for other radionuclides.

A mixture of radionuclides is classified as LSA if it satisfies the following condition (known as the ratio rule):

$$(ACT1/0.0001) + (ACT2/0.005) + (ACT3/0.03) < 1.0 \ , \quad (8)$$

where

ACT1 = total activity, in mCi/g of material, of all nuclides present with an A2 value of <0.05 Ci

ACT2 = total activity (mCi/g of material) of all nuclides present with an A2 value >0.05 but <1.0 Ci

ACT3 = total activity (mCi/g of material) of all nuclides with an A2 value >1.0 Ci.

7. *A1 activity* is the maximum activity of special form radioactive material permitted in a type A package. Values of A1 for all radionuclides are given in Appendix A of 10CFR71.

8. *A2 activity* is the maximum activity of radioactive material, other than special form radioactive material, permitted in a type A package. Values of A2 for all isotopes are also given in Appendix A of 10CFR71.

9. *Fissile material* (isotopes ^{233}U, ^{235}U, ^{238}Pu, ^{239}Pu, and ^{241}Pu) is placed into one of the following three classes, according to the controls needed to provide nuclear criticality safety during transportation:

a. Fissile Class I: a package that can be transported in unlimited numbers and in any arrangement and that requires no nuclear criticality safety controls during transportation. A transport index (see definition below) may be required because of external radiation levels.

b. Fissile Class II: a package that can be transported together with other packages in any arrangement, but for criticality control in numbers that do not exceed an aggregate transport index of 50. These shipments require no other nuclear criticality safety control during transportation. Individual packages may have a transport index not <0.1 and not >10.

c. Fissile Class III: a shipment of packages that is controlled in transportation by specific arrangements between the shipper and the carrier to provide nuclear criticality safety. (Paragraphs 10CFR71.18 to 71.24 define in detail the allowed composition of fissile classes II and III packages.)

10. *Transport index* is a dimensionless number, rounded off to the first decimal place, placed on the label of a package to designate the degree of control to be exercised by the carrier during transportation. It is defined as follows:

a. It is the number expressing the maximum radiation level in millirems per hour at 1 m from the external surface of the package.

b. For fissile class II packages, it is the number expressing the maximum radiation level in millirems per hour at 1 m from the external surface of the package, or the number obtained by dividing by 50 the allowed number of packages that may be transported together (as specified in 10CFR71.59), whichever number is larger.

[Transportation of fissile materials, particularly of classes II and III, requires special considerations (see 49CFR173.455). Transportation of plutonium is covered in Subtitle H, Section 180 of the NWPA Amendments.]

11. *Type A quantity* is the quantity of radioactive material, the aggregate of which does not exceed A1, for special form radioactive material, or A2, for normal form radioactive material, where A1 and A2 are specified in Appendix A of 10CFR71. For a mixture of radionuclides, rules given in 49CFR173.433 may be used. In general, the ratio rule is used that is analogous to Eq. (12) defining the LSA status of a mixture. In the present case, the activity of each nuclide is divided by its A1 or A2 value, as appropriate, and the resulting fractions are summed. If the sum is <1.0, the mixture is considered a type A quantity.

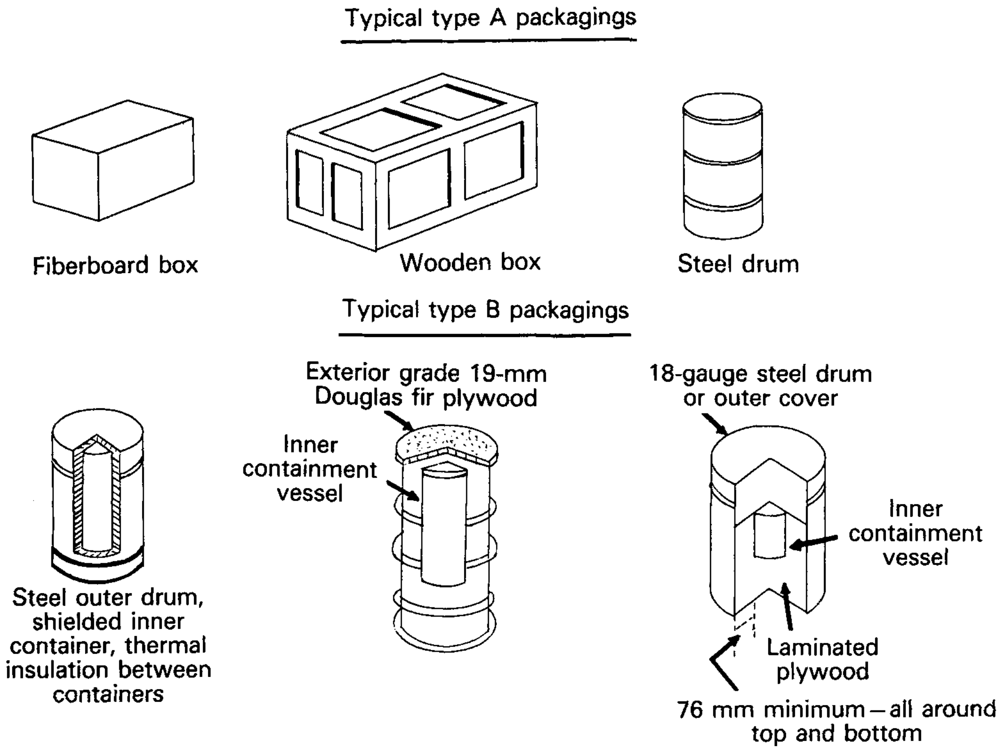


Fig. 25. Sketches of typical type A and B packages.

12. *Type B quantity* is a quantity of radioactive material greater than type A.

There is another class of materials, called mixed waste, that is regulated by the NRC and the EPA. Mixed waste is defined as waste that is considered radioactive (i.e., is subject to regulations applying to radioactive materials) and contains hazardous waste that either (a) is listed as hazardous waste in Subpart D of 40CFR261 (Title 40CFR contains EPA rules and regulations) or (b) exhibits any of the hazardous waste characteristics identified in Subpart C of 40CFR261. The hazardous component of mixed waste is regulated under the Resource Conservation and Recovery Act. Handling, packaging, and disposing of mixed waste requires special care and adherence to both NRC and EPA regulations.

Information about packages used for the transportation of radioactive materials is given in Refs. 50 and 83, particularly details about tests the packages must satisfy in order to qualify as shipping radioactive materials. In general, three types of containers are allowed: type A packages, type B packages, and "other." The requirement for type A or B container is based on the A1 and A2 activity values (see Table XIX).

A radionuclide or a mixture of radionuclides can be transported in a type A container, if its activity is type A quantity. A type B quantity must be transported in a type B container. Examples of such containers are shown in Fig. 25. A simple rule indicating the type of package required is shown in Fig. 26.

Radioactive materials that qualify as LSA can be transported in an "essentially" type A container, in the case of "nonexclusive use" transport medium. Essentially, type A means that the container does not have to satisfy all the tests of type A containers. The LSA materials transported by conveyances assigned for

	Activity < A1	Activity > A1
LSA	Activity < A1 ↓ Strong container	Activity > A1 ↓ Type A container
Non-LSA	Activity < A1 ↓ Type A container	Activity > A1 ↓ Type B container

Fig. 26. Simple chart for the selection of the container based on the activity of the package.

"exclusive use" may be shipped in packages of less rigorous construction. In this case, there should be an assurance that the package will not be loaded or unloaded except under the direction of the consignee or the consignor.

For the transportation of TRU waste, the DOE has sponsored the development of a special carrier, called TRUPACT (for TRU PAckage Transporter). The components of TRUPACT are shown in Fig. 27. The NRC announced in 1989 that it approved the design of TRUPACT-II (Ref. 86) for the shipment of TRU waste to WIPP. The dose rate requirements at the outside of packages and vehicles carrying radioactive packages are shown in Fig. 28.

In case of an accident involving radioactive materials in transit, there are procedures and guidelines developed by the Federal Emergency Management Agency (FEMA). The FEMA established and chairs the Federal Radiological Coordinating Committee and coordinates emergency management assistance from all federal agencies to state and local governments. There is a FEMA document that gives guidelines for state and local personnel with responsibility for emergency response plans.[87]

In addition to FEMA, the DOT has provided materials to states for training of fire, police, and ambulance personnel, and the DOE currently has eight regional offices for radiological assistance. Nationally, 28 DOE radiological assistance teams consisting of federal and contract employees are available. Finally, the Institute of Nuclear Power Operations has developed a voluntary assistance agreement among the electric utilities under which they will assist each other in cases of radioactive materials transportation accidents.

XVIII. CONCLUSIONS

In this paper, we have reviewed the past and present status of radioactive waste management. Sources and examples of radioactive materials with current classification schemes have been presented. The various policy acts of the federal government and their amendments that have been passed by the U.S. Congress have been enumerated. Additionally, the characteristics of HLW, TRU, and LLW have been discussed along with the currently planned and, in some cases, the implemented disposal methods both in the United States and abroad.

Current information on power reactor decommissioning has been presented and the recent NRC regulations for reactor decommissioning, including financing, have been discussed.

It is the authors' opinion that adequate government regulations and technical knowledge are available to manage the disposal of radioactive wastes safely and

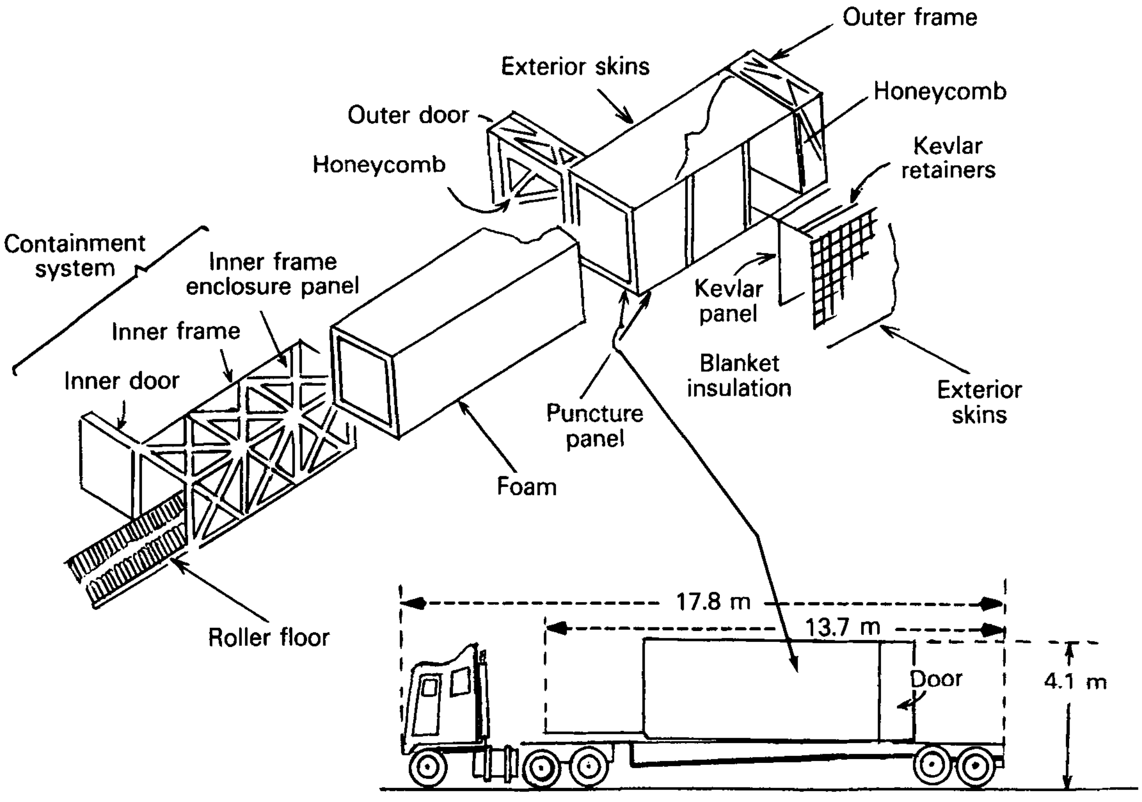


Fig. 27. Basic design of the TRUPACT carrier.

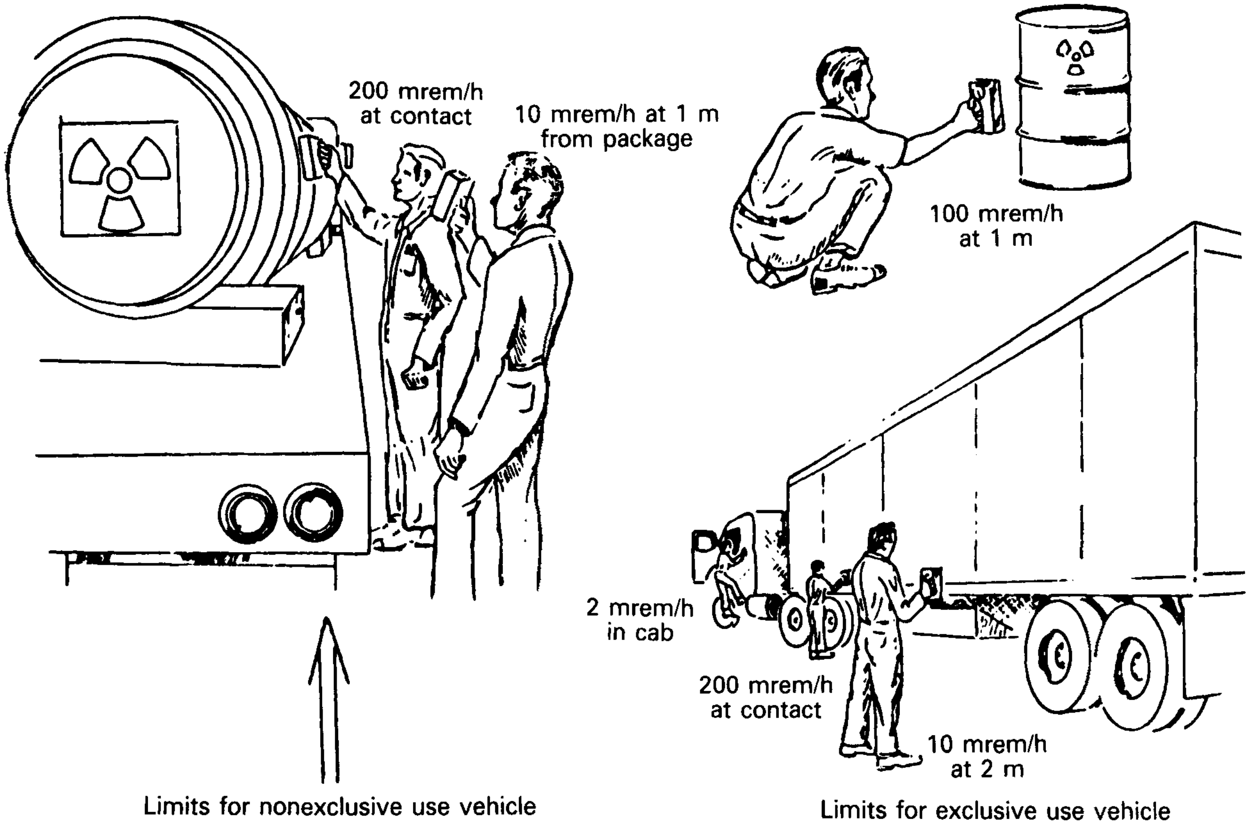


Fig. 28. Radiation dose limits for packages transported by exclusive and nonexclusive vehicles.

expeditiously. However, when one considers the political factors involved in the decision making process, the solution of this problem still faces many difficulties.

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